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DRAWINGS CITED IN THIS CHAPTER*

* 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 USAR. They are controlled by the Controlled Documents Program.

DRAWING *	SUBJECT
768E972	Group Classification & Containment Isolation Diagram
A21-1063	Typical Masonry Wall Details
A21-1065	Typical Shielding Wall Details
A26-1000-01A	Auxiliary Building Basement Plan Area 1
A26-1000-02A	Auxiliary Building Basement Plan Area 2
A26-1571	Auxiliary Building Wall Column Support Schedule
A26-1576	Auxiliary and Fuel Building Block Wall Tee Support Schedule
A28-1000-06A	Fuel Building Basement Plan Area 6
A28-1571	Fuel Building Block Wall Column Support Schedule
E27-1310	Electrical Penetrations
M01-1107	General Arrangement - Mezzanine Floor Plan El. 762'-0"
M01-1111	General Arrangement - Sections "C-C", "D-D" and "E-E"
M01-1600	Environmental Zone Map
M05-1073	Low Pressure Core Spray System
M05-10//74	High Pressure Core Spray System
M05-1075	Residual Heat Removal System
M05-1079	Reactor Core Isolation Cooling System
M27-1314	Reactor Head Piping Plan Elevation 804'-2-1/4"
S27-1401	Structural Instrumentation
W27-1000-00A	Containment Building El 712'-3", Auxiliary Building El 706'-6", 712'-0", Fuel Building El 712'-0" Masonry Wall Index Sheet

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

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 Summary Description

This section contains an evaluation of the design bases of the plant as measured against the NRC General Design Criteria for Nuclear Power Plants, Appendix A of 10 CFR 50, effective May 21, 1971, and subsequently amended February 20, 1976. The General Design Criteria, which are divided into six groups and total 55 in number, are intended to establish minimum requirements for the design of nuclear power plants.

It should be noted that the General Design Criteria were not written specifically for the BWR; rather, they were intended to guide the design of all water-cooled nuclear power plants. As a result, the criteria are generic in nature and subject to a variety of interpretations. For this reason, there are some cases where conformance to a particular criterion is not directly measurable. In these cases, the conformance of plant design to the interpretation of the criterion is discussed. For each of the 55 criteria, a specific assessment of the plant design is made and a complete list of references included to identify where detailed design information pertinent to each criterion is treated in this document.

Based on the content herein, Exelon Generation Company, LLC concludes that the Clinton Power 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 Criterion 1 - Quality Standards and Records

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

3.1.2.1.1.1 Evaluation Against Criterion 1

Structures, systems, and components important to safety are listed in Table 3.2-1. The quality assurance program is described in Chapter 17 and is applied to the items as noted in this table. The intent of the quality assurance program is to assure sound engineering in all phases of design, construction and operation through conformity to regulatory requirements and design bases described in the license application. In addition, the program assures adherence to specified standards of workmanship and implementation of recognized codes and standards in fabrication, construction, maintenance, and modification. It also includes the observance of proper preoperational and operational testing and maintenance procedures as well as the documentation of the foregoing by keeping appropriate records. The quality assurance program at Clinton Power Station and its contractors is responsive to and satisfies the intent of the quality-related requirements of 10 CFR 50, including Appendix B.

Structures, systems, and components are first classified with respect to their location, service, and relationship to the safety function to be performed. Recognized codes and standards are applied to the equipment in these classifications as necessary to assure a quality level consistent with the required safety function.

Records are maintained which demonstrate that the requirements of the quality assurance program are being satisfied. These records show that appropriate codes, standards, and regulatory requirements are observed, specified materials are used, correct procedures are utilized, qualified personnel are provided, and that the finished parts and components meet the applicable specifications for safe and reliable operation. These records are available so that desired items of information are retrievable. These records will be maintained by or under the control of CPS throughout the life of the associated item.

The quality assurance programs of CPS and its contractors satisfy the requirements of Criterion 1.

For further discussion, see the following sections:

a. Principal Design Criteria, 1.2;

b. Plant Description, 1.2;

c. Classification of Structures, Components, and Systems, 3.2; and

d. Quality Assurance, 17.

3.1.2.1.2 Criterion 2 - Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect:

a. appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated,

b. appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and

c. the importance of the safety functions to be performed.

3.1.2.1.2.1 Evaluation Against Criterion 2

The design criteria adopted for structures, systems and components depend on the magnitude and the probability of occurrence of natural phenomena at the specific site. The designs are based on the most severe of the natural phenomena recorded for the site with an appropriate margin to account for uncertainties in the historical data. Detailed discussion of the various phenomena considered and the design criteria developed are presented in the FSAR sections listed below.

The design criteria developed meet the requirements of Criterion 2.

For further discussion, see the following sections:

a. Meteorology, 2.3;

b. Hydrologic Engineering, 2.4;

c. Geology, Seismology and Geotechnical Engineering, 2.5;

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

e. Wind and Tornado Loadings, 3.3;

f. Water Level (Flood) Design, 3.4;

g. Missile Protection, 3.5;

h. Seismic Design, 3.7;

i. Design of Category 1 Structures, 3.8;

j. Mechanical Systems and Components, 3.9;

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

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

3.1.2.1.3 Criterion 3 - Fire Protection

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

3.1.2.1.3.1 Evaluation Against Criterion 3

Structures, systems, and components important to safety have been designed to meet the requirements of Criterion 3. Fire protection systems meeting the requirements of Criterion 3 have been provided.

The plant is designed to minimize the probability and effect of fires. Noncombustible and fire-resistant materials are used in the containment, control room, components of safety features systems, and throughout the unit wherever practical to reduce fire potential. Equipment and facilities for fire protection, including detection, alarms and extinguishment, are provided to protect both plant equipment and personnel from fire and the resultant release of toxic vapors. Release of toxic vapors from cabling is minimized by the selection of low-halide cables (i.e., bromide in lieu of chloride). Both automatic and manual types of fire-fighting equipment are provided.

Fire protection is provided by automatic deluge, water spray, sprinkler, Halon 1301, carbon dioxide, manual hose stations, and portable extinguishers, depending on the location and type of fire.

Fire-fighting systems have been designed to assure that their rupture or inadvertent operation will not significantly impair systems important to safety.

The fire-protection system consists of a reliable system, designed and installed in accordance with the requirements of the National Fire Protection Association, the Nuclear Energy Liability Property Insurance Association, and the applicable local codes and regulations.

The fire-protection system has been provided with test valves and facilities for periodic testing. Equipment is accessible for periodic inspection.

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

The NSSS portion of the plant uses noncombustible and heat-resistant materials wherever practical (metal cabinets, metal wireways, high melting insulation, etc). Cabling is suitably rated, and the cable tray loading is designed to minimize internal heat buildup. In addition, NSSS mechanical and electrical equipment is designed and tested to anticipated environmental conditions

The fire-protection system is discussed in detail in Subsection 9.5.1 and in the Clinton Power Station Fire Protection Evaluation Report, dated August 23, 1985.

3.1.2.1.4 Criterion 4 - Environmental and Missiles Design Bases

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

3.1.2.1.4.1 Evaluation Against Criterion 4

Structures, systems and components important to safety are designed to accommodate the effects of, and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents.

These structures, systems and components are appropriately protected against dynamic effects including the effects of missiles, pipe whipping and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.

The electrical equipment, instrumentation and associated cables of protection and engineered safety features systems which are located inside the containment are discussed in the FSAR sections listed below indicating the design requirements in terms of the time which each must survive the extreme environmental conditions following a loss-of-coolant accident.

The design of these structures, systems and components meets the requirements of Criterion 4.

For further discussion, see the following Sections:

a. Meteorology, 2.3;

b. Hydrologic Engineering, 2.4;

c. Geology, Seismology, and Geotechnical Engineering, 2.5;

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

e. Wind and Tornado Loading, 3.3;

f. Water Level (Flood) Design, 3.4;

g. Missile Protection, 3.5;

h. Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping, 3.6;

i. Seismic Design, 3.7;

j. Design of Seismic Category I Structures, 3.8:

k. Mechanical Systems and Components, 3.9;

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

m. Environmental Qualification of Electrical Equipment, 3.11;

n. Integrity of Reactor Coolant Pressure Boundary, 5.2;

o. Engineered Safety Features, 6;

p. Instrumentation and Controls, 7; and

q. Electric Power, 8.

3.1.2.1.5 Criterion 5 - Sharing of Structures, Systems, and Components

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

3.1.2.1.5.1 Evaluation Against Criterion 5

Criterion 5 is not applicable to Clinton since Clinton is a one-unit facility.

3.1.2.2 Group II - Protection by Multiple Fission Product Barriers

3.1.2.2.1 Criterion 10 - Reactor Design

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

3.1.2.2.1.1 Evaluation Against Criterion 10

The reactor core components consist of fuel assemblies, control rods, in-core ion chambers, neutron sources (when installed), and related items. The mechanical design is based on conservative application of stress limits, operating experience and experimental test results. The fuel is designed to provide high integrity over a complete range of power levels, including transient conditions. The core is sized with sufficient heat transfer area and coolant flow to ensure that fuel design limits are not exceeded under normal conditions or anticipated operational occurrences.

The reactor protection system is designed to monitor certain reactor parameters, sense abnormalities, and to scram the reactor thereby preventing fuel design limits from being exceeded 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 four independent battery backed non-interruptible power sources with two AC sources on the scram solenoids.

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

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

For further discussion, see the following sections:

a. principal design criteria, 1.2;

b. plant description, 1.2;

c. fuel mechanical design, 4.2;

d. nuclear design, 4.3;

e. thermal and hydraulic design, 4.4;

f. reactor recirculation system, 5.4;

g. reactor core isolation cooling system, 5.4;

h. residual heat removal system, 5.4; and

i. accident analysis, 15.

3.1.2.2.2 Criterion 11 - Reactor Inherent Protection

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

3.1.2.2.2.1 Evaluation Against Criterion 11

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. The boiling water reactor has an inherently large moderator-to-Doppler coefficient ratio which permits 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. deleted

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:

a. principal design criteria, 1.2;

b. nuclear design, 4.3; and

c. thermal and hydraulic design, 4.4.

3.1.2.2.3 Criterion 12 - Suppression of Reactor Power Oscillations

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

3.1.2.2.3.1 Evaluation Against Criterion 12

The reactor core is designed to ensure that no power oscillation will cause fuel design limits to be exceeded. The power reactivity coefficient is the composite simultaneous effect of the fuel temperature or Doppler coefficient, moderator void coefficient, and moderator temperature coefficient to the change in power level. It is negative and well within the range required for adequate damping of power and spatial xenon disturbances. Analytical studies indicate that for large boiling water reactors, underdamped, unacceptable power distribution behavior could be expected to occur only with power coefficients more positive than about -0.01 $\Delta k/k \ \Delta P/P$. Operating experience has shown large boiling water reactors to be inherently stable against xenon-included power instability. The large negative operating coefficients provide:

a. deleted

b. deleted

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 abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through an automatic scram. High integrity of the protection system is achieved through the combination of logic arrangement, trip channel redundance, power supply redundancy, and physical separation.

The reactor core and associated coolant, control and protection systems are designed to suppress any power oscillations which could result in exceeding fuel design limits. These systems assure that Criterion 12 is met.

For further discussion see the following sections:

a. principal design criteria, 1.2;

b. nuclear design, 4.3;

c. thermal and hydraulic design, 4.4;

d. reactor manual control system, 7.7; and

e. accident analysis, 15.

3.1.2.2.4 Criterion 13 - Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

3.1.2.2.4.1 Evaluation Against Criterion 13

The neutron flux in the reactor core is monitored by five subsystems. The source range monitor (SRM) subsystem measures the flux from startup through criticality. The intermediate range monitor (IRM) subsystem overlaps the SRM subsystem and extends well into the power range. The power range is monitored by many detectors which make up the local power range monitor (LPRM) subsystem. The output from these detectors is used in many ways. The output of selected, core-wide sets of detectors is averaged to provide a core average neutron flux. This output is called the average power range monitor (APRM) subsystem. The traversing incore probe (TIP) subsystem provides a means for calibrating the LPRM system. Both the IRM and APRM subsystems generate scram trips to the reactor trip system. All subsystems but the TIP subsystem generate rod-block trips. Additional information on the neutron monitoring system is given in Chapter 7.

The reactor protection system protects the fuel barriers and the nuclear process barrier by monitoring plant parameters and causing a reactor scram when predetermined set points 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 are classified as identified and unidentified, which corresponds respectively to the flow to the 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. High pump fillup rate and pumpout rate are alarmed in the main control room. 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 pre-established limits are exceeded.

As noted above, 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 abnormal operational occurrence or accident.

3.1.2.2.5 Criterion 14 - Reactor Coolant Pressure Boundary

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

3.1.2.2.5.1 Evaluation Against Criterion 14

The piping and equipment pressure parts within the reactor coolant pressure boundary through the outer isolation valve(s) are designed, fabricated, erected, and tested to provide a high degree of integrity throughout the plant lifetime. Section 3.2 classifies systems and components within the reactor coolant pressure boundary as Quality Group A. The design requirements and codes and standards applied to this quality group ensure a quality consistent with regulations.

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 ASME Boiler and Pressure Vessel Code, Section III, where applicable. Where reactor coolant pressure boundary piping penetrates the containment, the fracture toughness temperature requirements of the reactor coolant pressure boundary materials apply.

Piping and equipment pressure parts of the reactor coolant pressure boundary are assembled and erected by welding unless applicable codes permit flanged or screwed joints. Welding procedures are employed which produce welds of complete fusion and free of unacceptable defects. Welding procedures, welders, and welding machine operators employed in producing pressure-containing welds 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 qualification 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, Subsection 3.1.2.4.1.

Reactor coolant system pressure isolation valves are tested for leakage in accordance with the CPS Technical Specifications. See Subsection 3.9.6.4.

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 sections:

a. Principal Design Criteria, 1.2.1;

b. Design of Structures, Components, Equipment and Systems, 3;

c. Overpressurization Protection, 5.2;

d. Reactor Vessel, 5.3;

e. Component and Subsystem Design, 5.4;

f. Accident Analysis, 15; and

g. Quality Assurance, 17.

3.1.2.2.6 Criterion 15 - Reactor Coolant System Design

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

3.1.2.2.6.1 Evaluation Against Criterion 15

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, and the residual heat removal system. The 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 ASME Boiler and Pressure Vessel Code, Section III as indicated in Section 3.2,

The auxiliary, control and protection systems associated with the reactor coolant system act to provide sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. 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 assure 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 overpressure 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 provide assurance 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 portions of the USAR:

a. principal design criteria, 1.2;

b. design of structures, components, equipment and systems, 3;

c. overpressurization protection, 5.2.2;

d. reactor coolant pressure boundary leakage detection system, 5.2.5;

e. reactor vessel, 5.3;

f. reactor recirculation system, 5.4; and

g. accident analysis, 15.

3.1.2.2.7 Criterion 16 - Containment Design

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

3.1.2.2.7.1 Evaluation Against Criterion 16

The containment system consists of the following major components:

a. A drywell enclosing the reactor pressure vessel, the reactor coolant recirculation loops and pumps and other branch connections of the reactor primary coolant system. The drywell is a cylindrical reinforced concrete structure with a removable steel head.

b. A suppression pool containing a large amount of water used to rapidly condense steam from a reactor vessel blowdown or from a break in a major pipe.

c. A leaktight containment surrounding the drywell and the suppression pool.

d. A secondary containment containing a negative pressure plenum completely enclosing the containment building.

The drywell, suppression pool and containment are designed to condense the steam and contain fission product releases from the postulated design bases accident, i.e., the double-ended rupture of the largest pipe in the primary coolant system. The leaktight containment prevents the release of fission products to the environment. The containment building provides direct radiation shielding to protect operating personnel and/or the public. The secondary containment negative pressure plenum provides a controlled environment for collecting and filtering leakage from the primary containment prior to releasing to the atmosphere in a controlled manner.

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

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

For further discussion, see the following sections:

a. General Plant Description, 1.2;

b. Concrete Containment, 3.8.1;

c. Containment Systems, 6.2; and

d. Accident Analyses, 15.

3.1.2.2.8 Criterion 17 - Electrical Power Systems

An onsite electrical power system and an off site electrical power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that

a. specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences, and

b. the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electrical power supplies, including the batteries and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

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

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

3.1.2.2.8.1 Evaluation Against Criterion 17

Onsite and offsite electric power systems are provided for the Clinton Power Station to permit the functioning of structures, systems and components important to safety.

The Class 1E electric power system for the CPS unit consists of three electrically and physically independent distribution divisions. The onsite power supplies for each of these three divisions consist of a diesel generator and a battery supply. (A fourth Class 1E battery supply is utilized for portions of the emergency core cooling and reactor protection systems.) Class 1E loads with redundant safety functions are assigned to redundant divisions. The redundancy of the Class 1E load functions is such that the nuclear safety of the station is not degraded when electrical power is lost to one division due to a single failure.

The three redundant distribution divisions which comprise the onsite electric power system for Unit 1 are supplied with electric power from the transmission network via two physically independent circuits. One of these offsite sources is a 345-kV circuit from the switchyard through Reserve Auxiliary Transformer B, and the other is a 138-kV circuit from the Illinois Power Company grid system through the Emergency Reserve Auxiliary Transformer.

The 138-kV and 345-kV transmission lines used as the offsite power supplies are on physically separate rights-of-way and are electrically independent.

The electric power systems as designed meet the requirements of Criterion 17. For further discussion, see the following sections:

a. General Plant Description, 1.2;

b. Seismic Qualification of Seismic Category I Instrumentation and Electric Equipment, 3.10;

c. Environmental Design of Mechanical and Electrical Equipment, 3.11;

d. Offsite Power System, 8.2;

e. Onsite A-C Power Systems, 8.3.1; and

f. Onsite D-C Power Systems, 8.3.2.

3.1.2.2.9 Criterion 18 - Inspection and Testing of Electric Power Systems

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to asses the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically the following;

a. the operability and functional performance of the components of the systems such as onsite power sources, relays, switches, and buses; and

b. the operability of the systems as a whole and, under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system and the onsite power system.

3.1.2.2.9.1 Evaluation Against Criterion 18

The electric power systems important to nuclear safety are the Class 1E power systems.

The electrical equipment which comprises the Class 1E power systems (diesel generators, switchgears, motor control centers, etc.) is located so that it is easily accessible for inspecting the conditions of its components and items such as wiring, insulation, connections and terminal boards.

Periodic operability and functional performance tests can be made of major components of the Class 1E power systems. In addition, system operability can be tested periodically under conditions simulating design conditions (not including design-basis environmental conditions), including simulation of design-basis accident signals and transfer from offsite to onsite power sources on loss of all offsite power.

The design of the electric power systems important to nuclear safety provides inspection and testing in accordance with the requirements of Criterion 18.

For further discussion, see the following sections:

a. Onsite A-C power Systems, 8.3.1; and

b. Onsite D-C Power Systems, 8.3.2.

3.1.2.2.10 Criterion 19 - Control Room

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

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

3.1.2.2.10.1 Evaluation Against Criterion 19

The control room and its postaccident ventilation systems have been designed to satisfy Seismic Category I requirements, as discussed in Subsection 3.2.1. Under accident conditions, sufficient shielding and ventilation are provided to permit occupancy and access to the control room without receiving more than 5 rem whole body or 30 rem thyroid. Shielding is described in Subsection 12.3.2; ventilation is discussed in Subsection 9.4.1; accident analyses are discussed in Chapter 15; and habitability is discussed in Section 6.4.

The reactor plant can be brought to cold shutdown from outside the control room, as discussed in Subsection 7.4.1.4.

3.1.2.3 Group III - Protection and Reactivity Control Systems

3.1.2.3.1 Criterion 20 - Protection System Functions

The protection system shall be designed

a. to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and

b. to sense accident conditions and to initiate the operation of systems and components important to safety.

3.1.2.3.1.1 Evaluation Against Criterion 20

The reactor protection systems are the aggregate of protection systems or safety systems, including the reactor trip system, which are provided to sense abnormal and accident conditions and automatically initiate reactor shutdown and the operation of the other systems and components important to safety. The reactor trip 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 pre-established limits of anticipated operational occurrences. Scram trip settings are selected and verified to be far enough above or below operating levels to provide proper protection but not be subject to spurious scrams. The reactor trip system includes the power supplies, sensors, transmitters, 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, main steamline isolation valve closure and reactor vessel low and high water level will prevent fuel damage following abnormal operational transients. Specifically, these process parameters initiate a scram in time to prevent the core from exceeding thermal-hydraulic safety limits during abnormal operational transients. Additional scram trips are initiated by drywell high pressure and scram discharge volume high water level. Response by the reactor trip system is prompt, and the total scram time is short. Control rod scram motion starts in less than 180 msec after the sensor contacts actuate.

In addition to the reactor trip system, which provides for automatic shutdown of the reactor to prevent fuel damage, other protection systems are provided to sense accident conditions and initiate automatically the operation of other safety systems and safety components. 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.

The design of the protection systems satisfy the functional requirements as specified in Criterion 20.

For further discussion, see the following sections:

a. principal design criteria, 1.2;

b. reactivity control mechanical design, 4.6;

c. control rod drive housing supports, 4.6;

d. overpressurization protection, 5.2;

e. main steamline isolation valves, 5.4;

f. emergency core cooling system, 6.3;

g. reactor protection system, 7.2;

h. containment and reactor vessel isolation control system, 7.3;

i. emergency core cooling systems - instrumentation and control, 7.3;

j. neutron monitoring system, 7.6;

k. process radiation monitoring system, 7.6;

l. leak detection system, 7.6; and

m. accident analysis, 15.

3.1.2.3.2 Criterion 21 - Protection System Reliability and Testability

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that

a. no single failure results in loss of the protection function, and

b. removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated.

The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

3.1.2.3.2.1 Evaluation Against Criterion 21

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

The reactor trip 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 (trip) system initiates an automatic reactor shutdown if the monitored plant variables exceed pre-established limits. This system is arranged with four trip channels. An automatic or manual trip in any two trip channels will result in a scram. This logic scheme is called a two-out-of-four arrangement. The reactor protection (trip) system can be tested during reactor operation. Manual scram testing is performed by operating one of the four manual scram controls. Operating one manual scram control tests one trip channel. The total test verifies the ability to de-energize the scram pilot valve solenoids. Indicating lights verify that the actuator contacts have opened. This capability for a thorough testing program significantly increases reliability.

Control rod drive operability can be tested during normal reactor operation. Drive position indicator and in-core neutron detectors are used to verify control rod movement. Each partially or fully withdrawn control rod can be inserted one notch and then returned to the original position without significantly perturbing the nuclear system at most power levels. 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 closed to 90% of full open position without affecting reactor operation. If reactor power is reduced sufficiently, the isolation valve may be fully closed. Valve leakage rates can be determined during refueling operations.

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. System design and operating procedures also permit testing the discharge valves to the reactor recirculation loops. The low-pressure coolant injection mode can be tested after reactor shutdown.

Each active component of the emergency core cooling systems provided to operate in a design-basis accident 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 systems satisfy the requirements specified in Criterion 21.

For further discussion, see the following sections:

a. principal design criteria, 1.2;

b. reactivity control system, 4.6;

c. main steamline isolation valves, 5.4.5;

d. residual heat removal system, 5.4.7;

e. containment systems, 6.2;

f. emergency core cooling systems, 6.3;

g. reactor protection system, 7.2;

h. engineered safety features systems, 7.3; and

i. accident analysis, 15.

3.1.2.3.3 Criterion 22 - Protection System Independence

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

3.1.2.3.3.1 Evaluation Against Criterion 22

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 will not interfere with the operation of that function. Wiring for the reactor protection (trip) system outside of control room enclosures is run in rigid metallic conduits. No other wiring is run in these conduits. The wires from duplicate sensors on a common process tap are run in separate conduits. The system sensors are electrically and physically separated. Only circuits of the same division may be run in the same conduit.

The reactor protection (trip) 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 input for each actuator logic. When an essential monitored variable exceeds its scram trip point, it is sensed by four independent sensors. An intentional bypass, maintenance operation, calibration operation, or test can result in bypass of a single division of sensors. This leaves three channels per monitored variable, each of which can initiate a scram. Only two actuator logics must trip to initiate a scram. The two-out-of-four arrangement thus assures that a scram will occur as each monitored variable exceeds its scram setting.

The protection systems meet the design requirements for functional and physical independence as specified in Criterion 22.

For further discussion, see the following sections:

a. principal design criteria, 1.2;

b. main steamline isolation system, 5.4.5;

c. residual heat removal system, 5.4.7;

d. emergency core cooling systems, 6.3;

e. reactor trip system, 7.2;

f. engineered safety feature system, 7.3; and

g. accident analysis, 15.

3.1.2.3.4 Criterion 23 - Protection System Failure Modes

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

3.1.2.3.4.1 Evaluation Against Criterion 23

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

The environmental conditions in which the instrumentation and equipment of the reactor protection (trip) system must operate were considered in establishing the component specifications. Instrumentation specifications are based on the worst expected ambient conditions in which the instruments must operate.

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

For further discussion, see the following sections:

a. principal design criteria, 1.2;

b. emergency core cooling systems, 6.3;

c. reactor protection system, 7.2; and

d. engineered safety feature systems, 7.3.

3.1.2.3.5 Criterion 24 - Separation of Protection and Control Systems

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

3.1.2.3.5.1 Evaluation Against Criterion 24

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 will not impair the functional ability of the isolation control system to respond to essential variables.

Process radiation monitoring is provided on lines that serve as discharge routes for radioactive materials from the containment.

Four instrumentation channels are used to prevent an inadvertent scram or 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/or isolation.

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

For further discussion, see the following sections:

a. Principal Design Criteria, 1.2.1;

b. Functional Design of Reactivity Control System, 4.6;

c. Emergency Core Cooling System, 6.3;

d. Reactor Protection System, 7.2;

e. Engineered Safety Features System, 7.3;

f. Neutron Monitoring System, 7.6.1.5;

g. Process Radiation Monitoring System, 7.6.1.2.

h. NSSS Leak Detection System, 7.6.1.4; and

i. Reactor Manual Control System, 7.7.1.2.

3.1.2.3.6 Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions

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

3.1.2.3.6.1 Evaluation Against Criterion 25

The reactor protection (trip) 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 will initiate an automatic scram and not impair the remaining variables from being monitored, and if one channel fails, the remaining portions of the reactor trip system shall function.

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

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

For further discussion, see the following sections:

a. principal design criteria, 1.2;

b. reactivity control system, 4.3;

c. nuclear design, 4.3;

d. thermal and hydraulic design, 4.4;

e. reactor protection system, 7.2;

f. rod control and information system, 7.7; and

g. accident analysis, 15.

3.1.2.3.7 Criterion 26 - Reactivity Control System Redundancy and Capability

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

3.1.2.3.7.1 Evaluation Against Criterion 26

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 (B4C) powder and/or hafnium. Positive insertion of these control rods is provided by means of the control rod drive hydraulic system.

The control rods are capable of reliably controlling reactivity changes during normal operation (e.g., power changes, power shaping, xenon burnout, normal startup and shutdown) via operator-controlled insertions and withdrawals. The control rods are also capable of maintaining the core within acceptable fuel design limits during anticipated operational occurrences via the automatic scram function. The unlikely occurrence of a limited number of stuck rods during a scram will not adversely affect the capability to maintain the core within fuel design limits.

The circuitry for manual insertion or withdrawal of control rods is completely independent of the circuitry for reactor scram. This separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry. 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 control rod system includes appropriate margin for malfunctions such as stuck rods in the highly unlikely event that they do occur. Control rod withdrawal sequences and patterns are selected prior to operation to achieve optimum core performance, and simultaneously, low individual rod worths. The operating procedures to accomplish such patterns are supplemented by the rod pattern control system, which prevents rod withdrawals yielding a rod worth greater than permitted by the preselected rod withdrawal pattern. Because of the carefully planned and regulated rod withdrawal sequence, prompt shutdown of the reactor can be achieved with the insertion of a small number of the many independent control rods. In the event that a reactor scram is necessary, the unlikely occurrence of a limited number of stuck rods will not hinder the capability of the control rod system to render the core subcritical.

The second independent reactivity control system is provided by the reactor coolant recirculation system. By varying reactor flow, it is possible to affect the type of reactivity changes necessary for planned, normal power changes (including xenon burnout). In the unlikely event that reactor flow is suddenly increased to its maximum value (pump runout), the core will not exceed fuel design limits because the power flow map defines the allowable initial operating states such that the pump runout will not violate these limits.

The control rod system is capable of holding the reactor core subcritical under cold conditions, even when the control rod of highest worth is assumed to be stuck in the fully withdrawn position. This shutdown capability of the control rod system is made possible by designing the fuel with burnable poison (Gd_2O_3) to control the high reactivity of fresh fuel. In addition, the standby liquid control system is available to add soluble boron to the core and render it subcritical, as discussed in Subsection 3.1.2.3.8.1.

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

For further discussion, see the following sections:

a. principal design criteria, 1.2;

b. reactivity control system, 4.3;

c. standby liquid control system - instrumentation and control, 7.4; and

d. rod control and information system, 7.7.

3.1.2.3.8 Criterion 27 - Combined Reactivity Control Systems Capability

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

3.1.2.3.8.1 Evaluation Against Criterion 27

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

In the very unlikely event that more than one control rod fails to insert, and the core cannot be maintained in a subcritical condition by control rods alone as the reactor is cooled down subsequent to initial shutdown, the standby liquid control system (SLCS) will be actuated to insert soluble boron into the reactor core. The SLCS has sufficient capacity to ensure that the reactor can always be maintained subcritical, hence only decay heat will be generated by the core, and that decay heat can be removed by the residual heat removal system, ensuring that the core will always be coolable.

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

For further discussion, see the following sections:

a. principal design criteria, 1.2;

b. reactivity control system, 4.3;

c. nuclear design, 4.3;

d. thermal and hydraulic design, 4.4;

e. reactor protection system, 7.2;

f. rod control and information system, 7.7; and

g. accident analysis, 15.

3.1.2.3.9 Criterion 28 - Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither

a. result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor

b. sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steamline rupture, changes in reactor coolant temperature and pressure, and cold water addition.

3.1.2.3.9.1 Evaluation Against Criterion 28

The control rod system design incorporates appropriate limits on the potential amount and rate of reactivity increase. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worths. The rod pattern control system prevents withdrawal other than by the preselected rod withdrawal pattern. The rod pattern control system 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 velocity to less than 3.11 ft/sec. The control rods can be positioned at 6-inch steps and have a normal nominal withdrawal speed of 3 in/sec.

The accident analysis (Chapter 15) 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 and its support structures or other reactor pressure vessel internals are maintained so that the capability to cool the core is not impaired for any of the postulated reactivity accidents described in the accident analysis.

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

For further discussion, see the following sections:

a. principal design criteria, 1.2;

b. control rod drive systems, 3.9.4;

c. reactor core support structures and internals mechanical design, 4.2;

d. reactivity control system, 4.1 and 4.6;

e. nuclear design, 4.3;

f. control rod drive housing supports, 4.6;

g. overpressurization protection, 5.2;

h. reactor vessel and appurtenances, 5.3;

i. main steamline flow restrictor, 5.4;

j. main steamline isolation valves, 5.4;

k. control rod and reactivity-instrumentation and controls, 7.6 and 7.7; and

l. accident analysis, 15.

3.1.2.3.10 Criterion 29 - Protection Against Anticipated Operational Occurrences

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

3.1.2.3.10.1 Evaluation Against Criterion 29

The high functional reliability of the reactor protection (trip) system and reactivity control system is achieved through the combination of logic arrangement, redundancy, physical and electrical independence, functional separation, fail-safe design, and inservice testability. These design features are discussed in detail in Criteria 21, 22, 23, 24, and 26.

An extremely high reliability of timely response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance. Active components can be tested or removed from service for maintenance during reactor operation without compromising the protection or reactivity control functions even in the event of a subsequent single failure. Safety-related components such as control rod drives, main steam isolation valves, residual heat removal pumps, etc., are tested during normal reactor operation. Functional testing and calibration schedules are developed using available failure rate data, reliability analyses, and operating experience. These schedules represent an optimization of protection and reactivity control system reliability by considering, 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.

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.

For further discussion, see the following sections:

a. principal design criteria, 1.2;

b. main steamline isolation valves system, 5.4.5;

c. residual heat removal system, 5.4.7;

d. containment systems, 6.2;

e. emergency core cooling system, 6.3;

f. reactor protection system, 7.2;

g. engineered safety features systems, 7.3; and

h. accident analysis, 15.

3.1.2.4 Group IV - Fluid Systems

3.1.2.4.1 Criterion 30 - Quality of Reactor Coolant Pressure Boundary

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

3.1.2.4.1.1 Evaluation Against Criterion 30

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 and Table 3.2-1. Further, product and process quality planning is provided as described in Chapter 17 to assure 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 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.

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

For further discussion, see the following sections:

a. principal design criteria, 1.2;

b. overpressurization protection, 5.2;

c. reactor coolant pressure boundary leakage detection system, 5.2;

d. reactor vessel and appurtenances, 5.3;

e. reactor recirculation system, 5.4;

f. reactor vessel instrumentation, 7.7:

g. leak detection system, 7.6; and

h. quality assurance, 17.

3.1.2.4.2 Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions:

a. the boundary behaves in a nonbrittle manner; and

b. the probability of rapidly propagating fracture is minimized.

The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining:

a. material properties,

b. the effects of irradiation on material properties,

c. residual, steady-state and transient stresses, and

d. size of flaws.

3.1.2.4.2.1 Evaluation Against Criterion 31

Brittle fracture control of pressure-retaining ferritic materials is provided to ensure protection against non-ductile fracture. To minimize the possibility of brittle fracture failure of the reactor pressure vessel, the reactor pressure vessel is designed to meet the requirements of ASME Code, Section III.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. The NDT temperature increases as a function of neutron exposure at integrated neutron exposures greater than about 1×10^{17} nvt with 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 up on the reactor vessel wall. This annular volume contains the core shroud, jet pump assemblies, and reactor coolant. Assuming plant operation at rated power, and availability of 100% for the plant lifetime, the neutron fluence at the inner surface of the vessel causes a slight shift in the transition temperature. Expected shifts in transition temperature during design life as a result of environmental conditions, such as neutron flux, are considered in the design. Operational limitations assume that NDT temperature shifts are accounted for in the reactor operation.

The reactor coolant pressure boundary is designed, maintained, and tested such that adequate assurance is provided that the boundary will behave in a non-brittle manner throughout the life of the plant. Therefore, the reactor coolant pressure boundary is in conformance with Criterion 31.

For further discussion, see the following sections:

a. design of structures, components, equipment and system, 3; and

b. integrity of reactor coolant pressure boundary, 5.2.

3.1.2.4.3 Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

3.1.2.4.3.1 Evaluation Against Criterion 32

The design of the reactor pressure vessel includes provisions for inservice inspection. Access is provided for examination of the pressure vessel, system piping, pumps, valves and components (including support and pressure-retaining bolting) within the reactor coolant pressure boundary. Removable insulation is also provided as necessary. The reactor coolant pressure boundary shall be inspected in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. Section 5.2.4 defines the Inservice Inspection Plan, access provisions, and areas of restricted access.

Reactor vessel material specimens which include the base metal, weld metal, and heat affected zone metal will be encapsulated within the reactor pressure vessel to permit periodic evaluation in accordance with requirements of 10 CFR 50 Appendix H.

For further discussion, see the following sections:

a. Design of Structures, Components, Equipment and Systems, 3;

b. Integrity of Reactor Coolant Pressure Boundary, 5.2;

c. Inservice Inspection and Testing of Reactor Coolant Pressure Boundary, 5.2;

d. Reactor Vessel, 5.3; and

e. Component and Subsystem Design, 5.4.

3.1.2.4.4 Criterion 33 - Reactor Coolant Makeup

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

3.1.2.4.4.1 Response to Criterion 33

Means are provided for detecting reactor coolant leakage. The leak detection system consists of sensors and instruments to detect, annunciate, and in some cases isolate the reactor coolant pressure boundary from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and 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 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 concomitant with a loss of feedwater supply, makeup capabilities are provided by the RCIC system.

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.

For further discussion, see the following sections:

a. detection of leakage through reactor coolant pressure boundary, 5.2.5;

b. reactor core isolation cooling system, 5.4.6; and

c. emergency core cooling system, 6.3.

3.1.2.4.5 Criterion 34 - Residual Heat Removal

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

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

3.1.2.4.5.1 Evaluation Against Criterion 34

Residual heat removal system provides the means to:

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

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.

Two independent loops are located in separate protected areas.

Both normal a-c power and the auxiliary onsite power system provide adequate power to operate all the auxiliary loads necessary for plant operation. The power sources for the plant auxiliary power system are sufficient in number, and of such electrical and physical independence, that no single probable event could interrupt all auxiliary power at one time.

The plant auxiliary buses supplying power to engineered safety features and reactor protection systems and those auxiliaries required for safe shutdown are connected by appropriate switching to standby diesel-driven generators located in the plant. Each power source, up to the point of its connection to the auxiliary power buses, is capable of complete and rapid isolation from any other source.

Loads important to plant operation and safety are split and diversified between switchgear sections, and means are provided for detection and isolation of system faults.

The plant layout is designed to effect physical separation of essential bus sections, standby generators, switchgear, interconnections, feeders, power centers, motor control centers, and other system components.

Full capacity standby diesel generators are provided to supply a source of electrical power which is self-contained within the plant and is not dependent on external sources of supply. The standby generators produce a-c power at a voltage and frequency compatible with the normal bus requirements for essential equipment within the plant. Each of the diesel generators has sufficient capacity to start and carry the essential loads it is expected to drive.

The residual heat removal system is adequate to remove residual heat from the reactor core to assure fuel and reactor coolant pressure boundary design limits are not exceeded. Redundant reactor coolant recirculation paths are available to and from the vessel and RHR system. 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.

For further discussion, see the following sections:

a. Section 5.4, Residual Heat Removal System

b. Section 6.3, Emergency Core Cooling Systems

c. Section 8.3, Onsite Power System

d. Section 9.2, Water Systems

e. Section 15.0, Accident Analyses

3.1.2.4.6 Criterion 35 - Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that:

a. fuel and cladding damage that could interfere with continued effective core cooling is prevented, and

b. cladding metal-water reaction is limited to negligible amounts.

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

3.1.2.4.6.1 Evaluation Against Criterion 35

The Emergency Core Cooling Systems (ECCS) consist of the following:

a. high-pressure core spray system (HPCS),

b. automatic depressurization system (ADS),

c. low-pressure core spray system (LPCS), and

d. 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, controls and instrumentation. The HPCS system is provided to assure 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. A source of water is available from either the RCIC 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 clad temperature. The automatic depressurization system uses several 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 centrifugal pump that can be powered by normal auxiliary power or the standby a-c power system; a spray sparger in the reactor vessel, above the core (separate from the HPCS sparger); piping and valves to convey water from the suppression pool to the sparger; and associated controls and instrumentation. In case of low water level in the 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. 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.

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 vessel pressure.

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 Subsection 6.3.3 is an analysis to show that the emergency core cooling systems conform to 10 CFR 50, Appendix K. 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, and

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 presented in the evaluation against Criterion 34.

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 systems, including their supply, meets the requirements of Criterion 35.

For further discussion, see the following sections:

a. residual heat removal system, 5.4.7;

b. emergency core cooling system, 6.3;

c. auxiliary power systems, 8.3;

d. water systems, 9.2; and

e. accident analysis, 15.

3.1.2.4.7 Criterion 36 - Inspection of Emergency Core Cooling System

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

3.1.2.4.7.1 Evaluation Against Criterion 36

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 has been provided for examination of nozzles. Removable insulation is provided on the emergency core cooling systems piping where required for access. Inspection of the emergency core cooling systems is in accordance with the intent of Section XI of the ASME B&PV Code. Section 5.2.4 discusses the inservice inspection plan, access provisions, and areas of restricted access.

Pumps, valves, piping, and other components outside the drywell can be visually inspected at any time. Components located inside the drywell can be inspected only when the plant is shut down and the drywell is open for access. Adequate space has been provided for inspection and maintenance of pumps, valves, and their associated bolting. Portions of the ECCS which are part of the reactor coolant pressure boundary are designed to specifications for inservice inspection permitting detection of defects which might affect cooling performance. The design of the reactor vessel and internals for inservice inspection and plant testing and inspection programs ensures that the requirements of Criterion 36 are met.

For further discussion, see the following sections:

a. Mechanical Systems and Components, 3.9;

b. Inservice Inspection of Class 2 and 3 Components, 6.6;

c. Reactor Vessel, 5.3; and

d. Emergency Core Cooling Systems, 6.3.

3.1.2.4.8 Criterion 37 - Testing of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure

a. the structural and leaktight integrity of its components,

b. the operability and performance of the active components of the system, and

c. the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

3.1.2.4.8.1 Evaluation Against Criterion 37

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 assure the structural and leaktight integrity of its components.

The HPCS, LPCS, LPCI and the ADS are designed to permit periodic testing to assure 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 Systems will all be 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. It is concluded that the requirements of Criterion 37 are met.

For further discussion, see emergency core cooling systems, 6.3.

3.1.2.4.9 Criterion 38 - Containment Heat Removal

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptable low levels.

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

3.1.2.4.9.1 Evaluation Against Criterion 38

The containment heat removal function is accomplished by the residual heat removal system (RHR). Following a loss-of-coolant accident, one or both of the following operating modes of the RHR system would be initiated:

a. Containment spray - Condenses steam within the containment.

b. Suppression pool cooling - Limits the temperature within the containment by removing heat from the suppression pool water via the RHR heat exchangers. Either or both redundant RHR heat exchangers can be manually activated.

The redundancy and capability of the offsite and onsite electrical power systems for the residual heat removal system are presented in the evaluation against Criterion 34.

For further discussion, see the following sections:

a. residual heat removal system, 5.4.7;

b. containment systems, 6.2;

c. standby a-c power supply and distribution, 8;

d. water systems, 9.2; and

e. accident analysis, 15.

3.1.2.4.10 Criterion 39 - Inspection of Containment Heat Removal System

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

3.1.2.4.10.1 Evaluation Against Criterion 39

The design of the containment heat removal system includes provisions for periodic inspections of active components and other important equipment. Pumps, valves, piping, and other components outside the drywell can be inspected at any time. Space has been provided outside the drywell for inspection and maintenance. The suppression pool is designed to permit periodic inspection.

The testing frequencies of most components will be correlated with the component inspection. This design meets the requirements of Criterion 39.

For further discussion, see the following sections:

a. Residual Heat Removal System, 5.4.7;

b. Containment Systems, 6.2;

c. Emergency Core Cooling Systems, 6.3;

d. Engineered Safety Features, 6; and

e. Water Systems, 9.2.

3.1.2.4.11 Criterion 40 - Testing of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure:

a. the structural and leaktight integrity of its components,

b. the operability and performance of the active components of the system, and

c. the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

3.1.2.4.11.1 Evaluation Against Criterion 40

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

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 will be operated periodically to verify operability. The cooling mode is not automatically initiated but operation of the components is periodically verified. The operation of associated cooling water systems is discussed in the response to Criterion 46. It is concluded that the requirements of Criterion 40 are met.

For further discussion, see the following:

a. Section 5.4, Residual Heat Removal System

b. Section 6.3, Emergency Core Cooling System

c. Section 8.3, Onsite Power System

d. CPS Technical Specifications

3.1.2.4.12 Criterion 41 - Containment Atmosphere Cleanup

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

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

3.1.2.4.12.1 Evaluation Against Criterion 41

As described in other sections (e.g. Section 9.4), fans and coolers are provided in the drywell and containment areas to maintain suitable temperature conditions and to provide thorough mixing of the containment atmosphere during normal operation.

Fission products and all other materials are confined in the containment. Leakage from the primary containment is to the secondary containment from which the standby gas treatment system draws air as described in Section 6.5. The standby gas treatment system meets Seismic Category I requirements.

The exhaust from the secondary containment is routed through the standby gas treatment system, consisting of HEPA and charcoal filters, to minimize the release of radioactivity to the environment during abnormal occurrences.

The exhaust air from the primary and secondary containments is continuously monitored for radioactivity and combustible gas concentration. The normal ventilation system will be shutdown and isolated if high activity levels occur.

A combustible gas control system is provided to control the concentration of combustible gases in the containment. The system has redundant equipment and power supplies and can be manually operated as required.

The standby gas treatment system also functions as a backup to the hydrogen recombiner and can filter air purged from the primary containment post-LOCA. Operation of the standby gas treatment system in the post-LOCA purge mode will be restricted to less than 6% hydrogen concentration.

The above described systems meet the requirements of Criterion 41. For further discussion, see the following sections:

a. General Plant Description, 1.2;

b. Containment Functional Design, 6.2.1;

c. Secondary Containment Functional Design, 6.2.3;

d. Combustible Gas Control in Containment, 6.2.5;

e. Instrumentation and Controls, Chapter 7;

f. Standby Gas Treatment System (SGTS), 6.5;

g. Air Conditioning, Heating, Cooling, and Ventilation Systems, 9.4;

h. Gaseous Waste Management Systems, 11.3;

i. Process and Effluent Radiological Monitoring and Sampling System, 11.5; and

j. Accident Analysis, Chapter 15.

3.1.2.4.13 Criterion 42 - Inspection of Containment Atmosphere Cleanup Systems

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

3.1.2.4.13.1 Evaluation Against Criterion 42

Most of the components of the containment cleanup systems are located in areas which are accessible and can be inspected during normal plant operation. The piping and isolation valves in the drywell can be inspected during shutdown and refueling.

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

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

a. General Plant Description, 1.2;

b. Containment Functional Design, 6.2.1;

c. Secondary Containment Functional Design, 6.2.3;

d. Combustible Gas Control in Containment, 6.2.5;

e. Instrumentation and Controls, Chapter 7;

f. Standby Gas Treatment System (SGTS), 6.5;

g. Air Conditioning, Heating, Cooling, and Ventilation Systems, 9.4;

h. Gaseous Waste Management System, 11.3;

i. Process and Effluent Radiological Monitoring and Sampling Systems, 11.5; and

j. Accident Analysis, Chapter 15.

3.1.2.4.14 Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as, fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole, and under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation or applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

3.1.2.4.14.1 Evaluation Against Criterion 43

The integrity of the containment atmosphere cleanup systems is verified by preoperational and inservice testing. Preoperational and operational testing (including HEPA filter and charcoal adsorber) for the standby gas treatment system is discussed in Subsection 6.2.3.4. Inspection and testing of the containment fission product removal and control systems are discussed in Section 6.5. Testability of the power sources is described in Chapter 8.

Preoperational and operational testing for the drywell purge system is discussed in Subsection 9.4.7.2.4. Preoperational and operational testing for the combustible gas control system is discussed in Subsection 6.2.5.1.4.

3.1.2.4.15 General Design Criterion 44 - Cooling Water

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

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

3.1.2.4.15.1 Evaluation Against Criterion 44

The shutdown service water system described in Subsection 9.2.1 fulfills the requirements of this design criterion. This system pumps water from the ultimate heat sink described in Subsection 9.2.5 through the station and back to the ultimate heat sink.

This water removes heat from structures and nuclear safety-related components during normal station operation and under accident conditions.

Three independent, redundant, subsystems are provided to assure that a single failure will not effect the ability of the system to safely shut down the plant. All equipment for this system is provided with power from either the onsite diesel generators or the offsite buses so that loss of either source of electrical supply will have no impact on the ability of the system to safely shut down the plant.

Two of the subsystems are interconnected, with normally closed isolation valves between them. Each subsystem is provided with adequate instrumentation to determine leakage and has the capability of isolating specific pieces of equipment.

3.1.2.4.16 General Design Criterion 45 - Inspection of Cooling Water Systems

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

3.1.2.4.16.1 Evaluation Against Criterion 45

The shutdown service water system is designed to permit periodic inspection and testing of the pumps, strainers, valves, instruments and heat exchangers in the system. All components of the system are readily accessible for visual inspection during normal station operation.

3.1.2.4.17 General Design Criterion 46 - Testing of Cooling Water System

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

3.1.2.4.17.1 Evaluation Against Criterion 46

The shutdown service water system is designed to permit periodic functional testing as described in Subsection 9.2.1.2.4. The system will be initially hydrostatically tested prior to startup.

Periodic tests will verify the structural and leaktight integrity of the system. Functional tests will also be performed to ensure that all components remain operable.

The capability of the system to transfer from offsite electrical supply to onsite electrical power will also be periodically tested. In addition, the full sequence bringing all required equipment into operation for reactor shutdown, loss of offsite electrical power and the postulated loss-of-coolant accident will be periodically tested under conditions as close to actual as possible. Subsection 9.2.1.2.4.provides a more detailed discussion of specific testing programs.

3.1.2.5 Group V - Reactor Containment

3.1.2.5.1 Criterion 50 - Containment Design Basis

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect considerations of

a. the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning,

b. the limited experience and experimental data available for defining accident phenomena and containment responses, and

c. the conservatism of the calculational model and input parameters.

3.1.2.5.1.1 Evaluation Against Criterion 50

The design of the containment is based on the natural phenomena postulated to occur at the site and on the peak transient pressures and temperatures that could occur due to any postulated loss-of-coolant accident (LOCA) as discussed in Subsection 6.2.1.1.3. The LOCA includes the worst single failure (which leads to maximum containment pressure and temperature). These conditions are coupled with the loss of offsite power and the partial loss of the redundant engineered safety features systems (minimum engineered safety features).

The maximum pressure and temperature reached in the drywell and containment during this worst case accident are shown in Chapter 15 to be well below the design pressure and temperature of the structures. This provides an adequate margin for uncertainties in potential energy sources.

The design of the containment system thus meets the requirements of Criterion 50.

For further discussion, see the following sections:

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

b. Wind and Tornado Loadings:, 3.3;

c. Missile Protection, 3.5;

d. Protection Against Dynamic Effects, Associated with the Postulated Rupture of Piping, 3.6;

e. Seismic Design, 3.7;

f. Design of Seismic Category I Structure, 3.8;

g. Containment Functional Design, 6.2.1;

h. Containment Heat Removal System, 6.2.2; and

i. Accident Analyses, 15.

3.1.2.5.2 Criterion 51 - Fracture Prevention of Containment Pressure Boundary

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions

a. its ferritic materials behave in a nonbrittle manner, and

b. the probability of rapidly propagating fracture is minimized.

The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining

a. material properties,

b. residual, steady-state, and transient stresses, and

c. size of flaws.

3.1.2.5.2.1 Evaluation Against Criterion 51

The containment vessel liner material (carbon steel) with a nominal section thickness greater than 5/8 inch has a nil ductility transition temperature of at least 30°F below the minimum service temperature. Containment ventilation systems maintain the containment temperature at a suitable level during a shutdown of the unit during cold weather.

The preoperational test program and the quality assurance program will ensure the integrity of the containment and its ability to meet all normal operating and accident requirements.

The containment design thus meets the requirements of Criterion 51.

For further discussion, see the following sections:

a. Design of Category I Structures, 3.8; and

b. Quality Assurance, 17.

3.1.2.5.3 Criterion 52 - Capability for Containment Leakage Rate Testing

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

3.1.2.5.3.1 Evaluation Against Criterion 52

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.

The testing provisions provided and the test program meet the requirements of Criterion 52.

For further discussion, see the following Subsection:

a. Containment Leakage Testing, 6.2.6

3.1.2.5.4 Criterion 53 - Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit

a. appropriate periodic inspection of all important areas, such as penetrations,

b. an appropriate surveillance program, and

c. periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

3.1.2.5.4.1 Evaluation Against Criterion 53

There are special provisions for conducting individual leakage rate tests on applicable penetrations. Penetrations will be visually inspected and pressure tested for leaktightness at periodic intervals.

The provisions made for penetration testing meet the requirements of Criterion 53.

For further discussion, see the following subsection:

a. Containment Leakage Testing 6.2.6.

3.1.2.5.5 Criterion 54 - Piping Systems Penetrating Containment

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

3.1.2.5.5.1 Evaluation Against Criterion 54

Piping systems penetrating containment are designed to provide the required isolation and containment capabilities with redundancy and reliability of operation as required. These piping systems are provided with test connections to allow periodic valve leakage tests to be performed. Isolation valve arrangements and actuation systems are discussed in Subsection 6.2.4.

The engineered safety features actuation system has provisions for testing isolation valve operability.

Conformance to Criterion 54 is further discussed in Subsections 3.1.2.5.6 (Criterion 55), 3.1.2.5.7 (Criterion 56), and 3.1.2.5.8 (Criterion 57).

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

3.1.2.5.6 Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

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

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

c. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment, or

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

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

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

3.1.2.5.6.1 Evaluation Against Criterion 55

The reactor coolant pressure boundary (as defined in 10 CFR 50, Section 50.2 (v)) consists of the reactor pressure-vessel, pressure-retaining appurtenances attached to the vessel, and 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 containment have suitable isolation valves capable of isolating the containment thereby precluding any significant release of radioactivity.

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

For further discussion, see the following:

a. Integrity of Reactor Coolant Pressure Boundary, 5.2;

b. Containment Isolation Systems, 6.2.4;

c. Instrumentation and Controls, 7;

d. Accident Analyses, 15; and

e. CPS Technical Specifications

3.1.2.5.7 Criterion 56 - Primary Containment Isolation

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

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

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

c. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment, or

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

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

3.1.2.5.7.1 Evaluation Against Criterion 56

Lines which penetrate the containment and communicate with the containment interior are provided with two isolation valves, one inside the containment and the other outside containment. These valves are either locked closed, automatic isolation valves, or a combination thereof.

The manner in which the containment isolation system meets this requirement is detailed in:

a. Containment Isolation Systems, 6.2.4;

b. Accident Analyses, 15; and

c. CPS Technical Specifications

3.1.2.5.8 Criterion 57 - Closed System Isolation Valves

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

3.1.2.5.8.1 Evaluation Against Criterion 57

Each line that penetrates containment and is not connected to the containment atmosphere and is not part of the reactor coolant pressure boundary has at least one isolation valve, either automatic, locked closed, or capable of remote manual operation, located outside the containment near the penetration.

Details demonstrating conformance with Criterion 57 are provided in the following section:

a. Containment Isolation Systems, 6.2.4.

3.1.2.6 Group VI - Fuel and Radioactivity Control

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

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

3.1.2.6.1.1 Evaluation Against Criterion 60

Waste handling systems have been incorporated in the plant design for processing and/or retention of radioactive wastes from normal plant operations, including anticipated operational occurrences, to ensure that the effluent releases to the environment are as low as reasonably achievable and within the limits of 10 CFR 20 and in compliance with 10 CFR 50, Appendix I.

The plant is also designed with provisions to prevent radioactivity releases during accidents from exceeding the limits of 10 CFR 100, or, for the accidents analyzed using Alternative Source Terms, the limits of 10 CFR 50.67.

The principal gaseous effluents from the plant during normal operation are the noncondensible gases from the air ejectors. These gases are exhausted through a low temperature off-gas treatment system that includes charcoal absorbers. The effluent from this system is continuously monitored. The system will be shut down and isolated in the event of abnormally high radioactivity levels in the effluents.

Ventilation air from the various plant areas is continuously monitored and, for containment areas, will be exhausted through HEPA and charcoal filters if radioactive material release rate limits are reached.

In the event of an accident inside containment, noncondensible gases are contained within the leaktight containment vessel. Release of these effluents will be by controlled purging of the containment in the event the combustible gas control system is not able to properly maintain the containment atmosphere. Exhaust is monitored and released in a controlled manner through HEPA and charcoal filters.

Liquid radioactive wastes are collected in waste collector tanks, treated through processing equipment, and then returned to the plant systems. Excess water is released in a controlled manner to the environment. All discharges to the environment are routed through monitors that continuously monitor and record the activity of the waste and provide an alarm to the operator in the unlikely event of high activity level.

Solid wastes including spent resins, filter sludges, and evaporator bottoms are solidified. Contaminated tools, equipment, filter cartridges, and clothing are compacted and shipped offsite in approved shipping containers.

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

For further discussion, see the following sections:

a. General Plant Description, 1.2;

b. Detection of Leakage through Reactor Coolant Pressure Boundary, 5.2.5;

c. Containment Systems, 6.2;

d. Radioactive Waste Management, 11; and

e. Accident Analysis, 15.

3.1.2.6.2 Criterion 61 - Fuel Storage and Handling and Radioactivity Control

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

3.1.2.6.2.1 Evaluation Against Criterion 61

3.1.2.6.2.1.1 New Fuel Storage

New fuel is placed in dry storage in the new fuel storage vault which is located inside the fuel building. The storage vault within the building provides adequate shielding for radiation protection. Storage racks preclude accidental criticality (see the evaluation against Criterion 62). The new fuel storage vault has a normally open drain and a cover to mitigate the potential of flooding the vault. The new fuel storage racks do not require any special inspection and testing for nuclear safety purposes.

3.1.2.6.2.1.2 Spent Fuel Handling and Storage

Irradiated fuel is also stored in the fuel building. No spent (irradiated) fuel is stored inside the containment during plant operation. The upper containment pool and fuel building spent fuel pool waters are circulated through the fuel pool cooling and cleanup system (FPC&C) to maintain pool water temperature, purity, water clarity, water level, and radioactivity concentration level within design limits. Storage racks preclude accidental criticality (see the evaluation against Criterion 62).

Reliable decay heat removal is provided by the closed loop FPC&C system. It consists of two circulating pumps, two heat exchangers, two filter-demineralizers, two skimmer surge tanks, and the required piping, valves, and instrumentation. Water is collected in the skimmer surge tanks from both pools, pumped through a heat exchanger and discharged to the bottom of each pool through diffusers. Prior to the heat exchanger, a percentage of the water flow (approximately 1000 gpm) is diverted for processing through the filter-demineralizers and returned upstream of the heat exchangers. Pool water temperature is maintained at or below 140°F when removing the design heat load from the pools with component cooling water (CCW). If it appears that the pool temperature will exceed 150°F, the shutdown service water (SSW) system can replace CCW as the FPC&C heat exchanger cooling medium to increase cooling capacity. If the reactor is in a cold shutdown condition and in the refueling mode, the RHR system may supplement the FPC&C system during abnormal heat loads.

Temperature, pressure, conductivity, and level and radiation monitors and alarms are provided to indicate various changes in fuel pool water. Fission product concentration in the pool water is minimized by use of the filter-demineralizer. This minimizes the release of radioisotopes from the pool surfaces to the fuel building environment.

No special tests are required because at least one pump, heat exchangers, and filter-demineralizer are normally in operation while fuel is stored in the pool. Duplicate units are operated periodically to handle abnormal heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation, and trouble alarms is adequate to verify system operability.

3.1.2.6.2.1.3 Radioactive Waste Systems

The radioactive waste systems provide all equipment necessary to collect, process, and prepare for disposal all radioactive liquids, gases, and solid waste (with the exception of spent fuel) produced as a result of reactor operation.

Liquid radwastes are classified, contained, and treated as high or low conductivity, chemical, detergent, sludges or concentrated wastes. Processing includes filtration, ion exchange, analysis, and dilution. Liquid wastes are also decanted, and sludge is accumulated for disposal as solid radwaste. Wet solid wastes are solidified and packaged in steel drums. Dry solid radwastes are packaged in shielded steel drums or other suitable containers. Gaseous radwastes are monitored, processed, recorded, and controlled so that radiation doses to persons outside the controlled area are below those allowed by applicable regulations.

Accessible portions of the fuel and radwaste buildings shall have sufficient shielding to maintain dose rates within the limits set forth in 10 CFR 20 and 10 CFR 50.

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

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

For further discussion, see the following sections:

a. Residual Heat Removal System, 5.4.7;

b. Containment Systems, 6.2;

c. New Fuel Storage, 9.1.1;

d. Spent Fuel Storage, 9.1.2;

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

f. Air Conditioning, Heating, Cooling, and Ventilating Systems, 9.4;

g. Radioactive Waste Management, 11; and

h. Radiation Protection, 12.

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

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

3.1.2.6.3.1 Evaluation Against Criterion 62

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 sufficient spacing and/or neutron-absorbing material between the assemblies to assure that the array when fully loaded is substantially subcritical. Racks are designed to allow top loading of fuel elements into individual fuel assembly positions only. The new and spent fuel racks are Seismic Category I components.

New fuel is placed in dry storage in the top-loaded new fuel storage vault. This vault contains a drain to prevent the accumulation of water. The new fuel storage vault racks (located inside the fuel building) 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 0.95 for new dry fuel. K_{eff} will not exceed 0.95 if the new fuel is flooded.

Fuel is stored under water in the fuel building spent fuel pool and cask storage pool. The racks in which spent fuel assemblies are placed are designed and arranged with neutron-absorbing material and geometric spacing to ensure subcriticality in the storage pool. 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. New fuel was temporarily stored dry in the spent fuel storage racks during initial plant fuel loading and was maintained at a subcritical multiplication factor K_{eff} of less than 0.98 under all conditions.

CPS compliance with NRC General Design Criterion 62 for the high density, spent fuel storage racks has been documented in Reference 1 and Reference 4 for fuel bundles having a U-235 enrichment of 3.25 weight percent or less, with pool temperatures between 68°F and 260°F (Q & R 220.55). Compliance with higher enriched fuel bundles at pool temperatures down to 40°F has been documented in References 2 and 3.

The fuel storage racks located in the upper containment building store fuel under water during normal and refueling operations. However, only new fuel may be stored in these fuel storage racks during normal operation. The racks are designed and arranged such that spent and new fuel is maintained at a subcritical multiplication factor K_{eff} of less than 0.95 under normal and abnormal conditions.

Used fuel is stored dry at the onsite Independent Spent Fuel Storage Installation (ISFSI) in HI-STORM FW System dry casks. Criticality is controlled by geometry and by utilizing neutron poison in the fuel basket of the Multi-Purpose Canister (MPC).

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.

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

For further discussion, see the following sections:

a. Fuel Storage and Handling, 9.1; and

b. Refueling Interlocks, Systems, Instrumentation and Controls, 7.6.1.1.

3.1.2.6.4 Criterion 63 - Monitoring Fuel and Waste Storage

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

3.1.2.6.4.1 Evaluation Against Criterion 63

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, low level in the fuel storage pool and continuous level reading of skimmer surge tank levels. System temperature is also continuously monitored and alarmed in the control room.

Area radiation monitors are provided to continuously monitor the background levels in key areas and to alarm in the main control room if abnormal levels are detected. These systems satisfy the requirements of Criterion 63. For further discussion, see the following sections:

a. Area Radiation and Airborne Radioactivity Instrumentation, 12.3.4;

b. All other Instrumentation Systems Required for Safety, 7.6;

c. Fuel Storage and Handling, 9.1;

d. Liquid Radwaste System, 11.2;

e. Gaseous Waste Management System, 11.3; and

f. Solid Waste Management System, 11.4.

3.1.2.6.5 Criterion 64 - Monitoring Radioactivity Releases

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

3.1.2.6.5.1 Evaluation Against Criterion 64

The reactor containment atmosphere is continuously monitored for radioactivity during normal operation with a multi-channel radiation monitor (see description of leak detection system, Subsection 7.6.1.4). Gross gamma radiation monitors are provided to monitor radiation in the drywell and in the containment atmosphere during and following postulated accidents (see Subsection 7.5.1.4).

Effluent discharge paths from spaces containing components for recirculation of loss-of-coolant accident fluids are continuously monitored during normal operations and during postulated accidents. During normal operating conditions, exhaust effluents of the containment and fuel building HVAC systems are monitored by process radiation monitors (see Section 11.5). During an accident, the exhaust effluent is monitored by the standby gas treatment system monitors (see Section 11.5).

Effluent discharge paths are provided with continuous process radiation monitors and grab sample provisions. The common HVAC exhaust and the SGTS exhaust are each monitored with 3-channel radioactivity monitors (see Section 11.5). The liquid effluent discharge paths are monitored with gross gamma monitors (see Section 11.5).

The plant environs are monitored under normal and accident conditions by suitably located environmental monitors and by routine sampling (see Offsite Dose Calculation Manual and the Environmental Report - Operating License Stage).

The process, effluent, and airborne radiological monitoring and sampling systems are described in Chapters 11 and 12.

3.1.3 References

1. Dalton, J., "Licensing Submittal Report for the Clinton Power Station, Unit 1, Spent Fuel Storage Racks", Nuclear Energy Services Document 81A0681, July 2, 1981.

2. "Clinton Power Station Fuel Storage k-infinity Conversion Analyses", General Electric Company Report GENE-155-93034, July 1993.

3. IP Nuclear Station Engineering Department Calculation IP-F-0096, "Nuclear Evaluation of High Density Spent Fuel Storage Racks at Low Temperature", CPS, Clinton, IL.

4. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment 170 for Clinton Power Station, October 31, 2005.

ATTACHMENT A3.1*

EXHIBIT A

LICENSING SUBMITTAL REPORT

FOR THE

CLINTON POWER STATION, UNIT 1

SPENT FUEL STORAGE RACKS

PREPARED FOR

ILLINOIS POWER COMPANY

* REMOVED BY USAR CHANGE PACKAGE 7-134.

REFER TO SECTION 3.1.3 REFERENCES.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Certain structures, components, and systems of the nuclear plant are considered important to safety because they perform safety actions required to avoid or 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 assure 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 necessary to assure:

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 comparable to the guideline exposures of 10 CFR 100, or, for the accidents analyzed using Alternative Source Terms, the limits of 10 CFR 50.67.

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

Components, equipment, and systems designated as Safety Class 1, Safety Class 2, or Safety Class 3 (see Subsection 3.2.3 for a discussion of safety classes) are classified as Seismic Category I except as noted in Table 3.2-1.

Seismic Category I structures, systems, and components have been analyzed under the loading conditions of the SSE and the 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 is based on levels of material stress or load factors, whichever is applicable, and yielded margins of safety appropriate for each earthquake. The margin of safety provided for safety class components, equipment, and systems for the SSE is sufficiently large to assure that their design functions are not jeopardized. For further details of seismic design criteria refer to the following portions of the USAR:

a. mechanical: Subsection 3.7.3;

b. electrical: Section 3.10;

c. structures: Section 3.7.2; and

d. instrumentation and controls: Section 3.10.

The OBE as defined in 10 CFR 100, Appendix A, is not incorporated as a part of the seismic classification scheme.

The seismic classification indicated in Table 3.2-1 meet the requirements of NRC Regulatory Guide 1.29, except for the clarification listed in Section 1.8.

Components (and their supporting structures) that are not Seismic Category I and whose collapse could result in loss of required function through impact or flooding of Seismic Category I structures, equipment, or systems required after a safe shutdown earthquake are analytically checked and designed to confirm their integrity against collapse when subjected to seismic loading resulting from the safe-shutdown earthquake. Exceptions include some safety-related cables and instrumentation that is located in the non-seismic Turbine Building. These cables and devices provide inputs to the solid-state protection system for reactor trip or perform functions initiated by the protection system. The principal function of these devices is to provide anticipatory trip for the reactor based upon secondary system parameters. If these cables and/or devices failed, other parameters not measured in the turbine building would provide the necessary signal to shut down the reactor.

Design of non-seismic Category I components and supports is accomplished by a design verification program to demonstrate that the nonseismically designed components located in Seismic Category I buildings will not collapse or lead during a dynamic event. (Non-Category I classified components are excluded from analysis provided they are identified as being located in areas that do not contain safety-related components.)

In addition, a seismic interaction program will ensure that no physical interaction between safety-related components and other plant components will prevent the safety-related component from fulfilling its safety-related function except as noted above for safety-related cable and instrumentation located in the Turbine Building.

1. Design Verification Program

   a. Nonseismically Supported Piping

      A verification by calculation has been performed to show that the design of the nonseismically supported piping and supports has sufficient margin of safety to sustain the dynamic loads and protect against pipe break, support failure, and overall collapse. Selected, representative nonseismically support subsystems have been analyzed for the combined loading of weight, thermal expansion, safe shutdown earthquake, and pool dynamic events. The results of this study show that under faulted conditions the nonsafety-related piping subsystems will not collapse or leak, the component support stresses will not exceed the ultimate strength of the material, and the auxiliary steel stresses will not exceed that allowed by NRC Standard Review Plan 3.8.3 Section II.5 (Reference NUREG 75/078 dated September 1975), for load combination number 6, as defined in Section II.3.

      This study was extended to include nonessential piping located in the pool swell region of the wetwell and nonessential piping located in the weir swell region of the drywell.

   b. HVAC Ducts and Hangers

HVAC ductwork and hangers in Category I buildings have been designed as Category I, except the HVAC ductwork and hangers located in the control building laboratory areas at Elevation 737' and the plenum located on top of the reactor pedestal which are designed as non-Category I because there are not Category I components in the laboratory and because the displacement of the plenum will not affect any Category I components.

c. HVAC Components and Instrumentation

The support systems for unit heaters, area coolers, duct accessories, HVAC instrument lines, instrumentation and panels located in Category I buildings have been designed for the seismic loads.

d. Non-Category I Equipment

Equipment foundations for non-Category I classified components have been designed for dynamic load. Equipment anchorage has been designed for faulted conditions.

e. Non-Category I Structural Items

Non-Category I structural components within Category I buildings like stairs and galleries have been designed so that they would not fall during a dynamic event. The supports for these items are designed for the dynamic loads, including safe shutdown earthquake and pool dynamic loads.

f. Non-Category I Cable Tray Supports, Conduit Supports, and Bus Duct Supports

In general, cable tray supports, conduit supports, bus duct supports and lighting system support attachments in Category I buildings are designed seismically. Those that were not designed seismically will be analyzed to verify that the ability of safety-related systems and components to perform their safety function will not be impaired by the failure of these supports.

The following components are not seismically designed:

a. Conduit in Control Room Elevator Machinery Room elevation 848’-2”

b. Conduit in lab area in Control Building, elev. 737’ with exception of Class 1E conduit C02401, C02402, C02502, and C02523 and pullboxes 1PB0273 and 1PB0405. Additions to this area should be evaluated for the effect on these Class 1E components.

c. Conduit in Control Building elevation 751’ in the area between S-T and 124-202.

g. Lighting and Communication Supports

Lighting fixtures/conduit and communication conduit are seismically supported in the Category I buildings.

2. Interaction Analysis Program

Class D piping and other non-safety-related components shall be evaluated to assure that any interaction with a safety-related component will not impair the capability of the safety-related component to perform its intended function.

A field verification by an area walkdown basis ensured adequate clearance exists around all safety-related components. Situations which violate the clearance criteria will be identified and resolved by either detailed calculations (to show that the components will not interact or that the interaction is not detrimental to the safety-related component) or by a redesign to eliminate the interaction. (MEB (DSER) 3)

3.2.2 System Quality Group Classifications

System quality group classifications as defined in NRC Regulatory Guide 1.26 have been determined for each water-, steam-, or radioactive-waste-containing component of those applicable fluid systems 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.

A tabulation of quality group classification for each component so defined is shown in Table 3.2-1 under the heading, "Quality Group Classification." Piping supports and pipe restraints for piping tabulated in Table 3.2-1 are of the same quality group as the piping. Drawing 768E972 is a diagram which depicts the relative locations of these components along with their quality group classification.

System quality group classifications and design and fabrication requirements as indicated in Tables 3.2-1 and 3.2-2 meet the requirements of Regulatory Guide 1.26.

3.2.3 System Safety Classifications

Systems and components are classified as Safety Class 1, Safety Class 2, Safety Class 3, or Other in accordance with the importance to nuclear safety. If the systems and components identified involve both mechanical and electrical components, then the safety class is stated for the mechanical components while the electrical components are classified in the electrical classification column per Subsection 3.2.3.4.3. Where the identified system and components have only electrical components, the safety class is indicated by an asterisk (*) and the electrical classification column is completed in accordance with Subsection 3.2.3.4.3. When the safety class column is not applicable, it is marked "N/A", as is the case of civil structures.

Equipment is assigned a specific safety class, recognizing that components within a system may be of differing safety importance. A single system may thus have components in more than one safety class. Piping and equipment supports are the same safety class as the piping or equipment that is supported.

The safety classes are defined in this section, and examples of their broad application are given. Because of specific design considerations, these general definitions are subject to interpretation and exceptions. Table 3.2-1 provides a summary of the safety classes for the principal systems and components of the plant.

Design requirements for components of safety classes are also delineated in this section (Table 3.2-2). Where possible, reference is made to accepted industry codes and standards (Table 3.2-3) which define design requirements commensurate with the safety function(s) to be performed. In cases where industry codes and standards have no specific design requirements, the locations of the appropriate subsections that summarize the requirements to be implemented in the design are indicated.

3.2.3.1 Safety Class 1

3.2.3.1.1 Definition of Safety Class 1

Safety Class 1, SC- 1, applies to components of the reactor coolant pressure boundary or core support structure whose failure could cause a loss of reactor coolant at a rate in excess of the normal makeup system.

3.2.3.2 Safety Class 2

3.2.3.2.1 Definition of Safety Class 2

Safety Class 2, SC- 2, applies to those systems, and components, other than service water systems, that are not Safety Class 1 but are necessary to accomplish the safety function of:

a. inserting negative reactivity to shut down the reactor,

b. preventing rapid insertion of positive reactivity,

c. maintaining core geometry appropriate to all plant process conditions,

d. providing emergency core cooling,

e. providing and maintaining containment isolation,

f. removing residual heat from the reactor and reactor core, and

g. storing spent fuel.

Safety Class 2 includes the following:

a. The reactor protection system.

b. Those components of the control rod system which are necessary to render the reactor subcritical.

c. Systems or components which restrict the rate of insertion of positive reactivity.

d. The assembly of components of the reactor core which maintain core geometry, including the fuel assemblies, core support structure, and core grid plate, as examples.

e. Other components within the reactor vessel such as jet pumps, core shroud, and core spray components which are necessary to accomplish the safety function of emergency core cooling.

f. Emergency core cooling systems.

g. Standby gas treatment system.

h. Postaccident containment heat removal systems.

i. Containment hydrogen recombiners.

j. Initiating systems required to accomplish safety function, including emergency core cooling initiating system and containment isolation initiating system.

k. At least one of the systems which recirculates reactor coolant to remove decay heat when the reactor is pressurized, and the system to remove decay heat when the reactor is not pressurized.

l. Spent fuel storage racks and spent fuel pools.

m. Reactor core isolation cooling system.

n. Electrical and instrument auxiliaries necessary to operate the above. Electrical components in this safety class are Class 1E and are indicated by an asterisk.

o. Pipes having a nominal pipe size of 3/4 inch or smaller, that are connected to the reactor coolant pressure boundary.

Systems and components in Safety Class 2 are listed in Table 3.2-1.

3.2.3.3 Safety Class 3

3.2.3.3.1 Definition of Safety Class 3

Safety Class, SC- 3, applies to those systems and components that are not Safety Class 1 or Safety Class 2, but

a. Whose function is to process radioactive fluids and whose postulated failure would result in conservatively calculated offsite doses that exceed 0.5 rem to the whole body or its equivalent to any part of the body in accordance with Regulatory Guide 1.26.

b. Which provide or support any safety system function.

Safety Class 3 includes the following:

a. Gaseous waste disposal system components, unless it can be demonstrated that a single component failure would not result in calculated potential offsite doses as mentioned in Subsection 3.2.3.3.1.a above.

b. Those portions of the radwaste equipment required to prevent leakage to the environs, at an excessive rate, of liquids from the liquid waste disposal system.

c. Service water systems required for the purpose of:

   1 Removal of decay heat from the reactor.

   2. Emergency core cooling.

   3. Post-accident heat removal from the suppression pool.

   4. Providing cooling water needed for the functioning of emergency systems.

d. Fuel supply for the onsite emergency electrical system.

e. Emergency equipment area cooling.

f. Compressed gas or hydraulic systems required to support control or operation of safety systems.

g. Electrical and instrumentation auxiliaries necessary for operation of the above. Electrical components within this safety class are Class 1E and are indicated by an asterisk.

3.2.3.4 Other Systems and Components

3.2.3.4.1 Definition of Other Systems and Components

A boiling water reactor has a number of 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 safety classes defined above. Such systems and components are designated as "Other".

3.2.3.4.2 Design Requirements for Other Systems and Components

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 will operate. Where possible, design requirements are based on applicable industry codes and standards. Where these are not available, the designer utilized accepted industry or engineering practice.

3.2.3.4.3 Electrical Classification

Structures, systems and components shall be classified as Class 1E, Non-Class 1E, or N/A (not applicable) in accordance with the following:

a. Class 1E

The safety classification of the electric equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment.

b. Non Class 1E

Non-Class 1E is the classification to be applied to all electric structures, systems and components other than Class 1E structures, systems and components.

c. N/A (Not Applicable) Classifications

N/A indicates that the electrical classification is not applicable to this structure, system or component. For example, an assembly (such as a water strainer) may have no electrical components.

3.2.4 Quality Assurance

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.

3.2.5 Correlation of Safety Classes with Industry Codes

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

Table 3.2-4 presents ASME Code, Section III edition and addenda that apply to Quality Group A components.

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
I. Reactor System (Sections 4.0 and 5.3) (v)							
1. Reactor vessel	1	I	A	B		D	N/A
2. Reactor vessel support skirt	1	I	A	B		D	N/A
3. Reactor vessel appurtenances, pressure retaining portions	1	I	A	B		D	N/A
4. CRD housing supports	2	I	N/A	B		D	N/A
5. Reactor internal structures, engineered safety features	2	I	N/A	B		D	N/A
6. Reactor internal structures, other	Other	N/A	N/A	N/A	(dd)	D	N/A
7. Control rods	2	I	N/A	B		D	N/A
8. Control rod drives	2	I	N/A	B		D	N/A
9. Core support structure	2	I	N/A	B		D	N/A
10. Fuel assemblies	2	I	B	B		D	N/A
11. Power range detector hardware	2	I	N/A	B		D	N/A
II. Nuclear Boiler System (Sections 5.2, 5.4.4, and 5.4.5) (v)							
1. Vessels, level instrumentation condensing chambers	1	I	A	B		D	N/A
2. Piping, relief valve discharge and vent including vacuum relief valves	3	I	C	B		C,D	N/A
3. Piping, main steam with-in outboard isolation valve	1	I	A	B		C,D	N/A
4. Piping, feedwater within outboard isolation valve	1	I	A	B		C,D	N/A
5. Piping, other within outboard isolation valves	1	I	A	B	(g)	C,D	N/A
6. Safety/relief valves	1	I	A	B		D	1E
7. Valves, main steam isolation valves	1	I	A	B		A,D	1E
8. Valves, other than isolation valves and within drywell	1	I	A	B	(g)	D	1E
9. Electrical modules with safety functions	*	I	N/A	B	(w)	N/A	1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

PRINCIPAL COMPONENTS(s)		SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
10.	Cables with safety function	*	I	N/A	B		N/A	1E
11.	Valves, feedwater isolation	1	I	A	B		D,A	1E
12.	Mechanical modules, instrumentation, with safety function	2	I	B	B		C	N/A
13.	Quencher, Discharge Device	3	I	C	B		C	N/A
14.	Main steam process piping from outboard isolation valve to shutoff valve, and feedwater piping between outboard (motor-operated and check) isolation valves	2	I	B	B		A	N/A
15.	Valves, shutoff	2	I	B	B		A	1E
16.	SRVM System	*	I	N/A	B		D,C,X	1E
III.	Recirculation System (Section 5.4.1) (v)							
1.	Piping	1	I	A	B	(g)	D	N/A
2.	Pumps	1	I	A	B	(g)	D	N/A
3.	Valves	1	I	A	B		D	non-1E
4.	Motor, pump	Other	N/A	N/A	N/A		D	non-1E
5.	Electrical modules with safety function	*	I	N/A	B	(w)	N/A	1E
6.	Cables with safety function	*	I	N/A	B		N/A	1E
7.	LFMG set	Other	N/A	N/A	N/A		F	non-1E
8.	RPT Circuit Breakers 3A, 3B, 4A, 4B	*	I	N/A	B		F	1E
9.	Recirc Pump Switchgear except RPT Circuit Breakers	Other	N/A	N/A	N/A		A	non-1E
10.	Hydraulic flow control equipment (FCV actuator, circulation unit, HPU, interconnecting piping/valves)	Other	N/A	D	N/A		C	N/A

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION (t)
IV. CRD Hydraulic System (Section 4.6.1) (v)							
1. Valves, scram discharge volume lines and containment isolation	2	I	B	B	(g)	C	N/A
2. Valves insert and withdrawal lines	2	I	B	B	(f)	C	N/A
3. Valves, other	Other	N/A	D	N/A	(g)	C,A	non-1E
4. Piping, scram discharge volume lines and containment penetration	2	I	B	B		C,A	N/A
5. Piping, insert and withdrawal lines	2	I	B	B	(f)	C,A	N/A
6. Piping, other	Other	N/A	D	N/A	(g)	C,A	N/A
7. Hydraulic control unit							
a. Scram solenoid valves, test switches, and associated wiring	2	I	Special	B	(k) (ee)	C	non-1E
b. All other electrical components	Other	I	Special	B	(k)	C	non-1E
8. Electric modules with safety function	*	I	N/A	B	(w)	N/A	1E
9. Cables with safety function	*	I	N/A	B		N/A	1E
10. CRD water pumps	Other	N/A	D	N/A		T	non-1E
V. Standby liquid control system							
1. Standby liquid control tank	2	I	B	B		C	N/A
2. Pump	2	I	B	B		C	N/A
3. Pump motor	Other	I	N/A	B		C	1E
4. Valves, explosive	2	I	A	B		C	non-1E
5. Vales, drywell, isolation and within	2	I	A	B	(g)	C	N/A
6. Valves, beyond drywell isolation valves	2	I	B	B	(g)	C	1E
7. Piping, downstream of explosive valves	1	I	A	B	(g)	C	N/A

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
8. Piping, upstream of explosive valves	2	I	B	B	(g)	C	N/A
9. Electrical modules with safety function	*	I	N/A	B	(w)	C	1E
10. Cables with a safety function	*	I	N/A	B		C	1E
VI. <u>Neutron Monitoring System</u> (Sections 7.1.2.1.4, 7.2, and 7.6.1.5) (v)							
1. Electric modules, IRM, APRM, OPRM	*	I	N/A	B	(w)	D	1E
2. Cable IRM, APRM	*	I	N/A	B		D	1E
3. Purge Valve	N/A	N/A	D	N/A		C	N/A
VII. <u>Reactor Protection</u> (Section 7.1.2.1.1 and 7.2) (v)							
1. Instrument inverters and distribution panels	*	I	N/A	B		X	1E
2. Cables with safety function	*	I	N/A	B		N/A	1E
3. Solenoid inverters, Regulating Transformers, EPA'S, and output breakers	*	I	N/A	B		A	1E
4. Solenoid inverters, except 3 above, and solenoid distribution panels	Other	I	N/A	(ee) B		A	non-1E
5. Electrical modules with safety function	*	I	N/A	B		N/A	1E
VIII. <u>Leak Detection System</u>							
1. Temperature element	2	I	N/A	B	(o) (w)	X	1E
2. Temperature and differential temperature switches	2	I	N/A	B	(o) (w)	A,C	1E
3. Differential flow switch	2	I	N/A	B	(o) (w)	A,C	1E
4. Differential pressure switch	2	I	N/A	B	(o) (w)	A,C	1E
5. Differential flow summer and square root converter	2	I	N/A	B	(w)	A,C	1E
6. Pressure transmitter and differential pressure transmitter	2	I	B	B	(o) (w)	S,A,F,X	1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
IX. Process Radiation Monitors							
1. Electric modules, containment building fuel transfer vent plenum, containment building exhaust, fuel building exhaust, and main control room intake monitors	*	I	N/A	B	(w)	N/A	1E
2. Cables with safety function	*	I	N/A	B		N/A	1E
3. Main steamline	Other	I	N/A	B	(w)	N/A	1E
X. RHR System (Sections 5.4.7, 6.2.2, 6.3, and 6.5.2) (v)							
1. Heat exchangers, shell side	2	I	B	B		A	N/A
2. Heat exchangers, tube side	3	I	C	B		A	N/A
3. Piping within outboard reactor coolant pressure boundary isolation valves	1	I	A	B	(g)	A,C	N/A
4. Piping, beyond outboard reactor coolant pressure boundary isolation valves	2	I	B	B	(g)	A,C	N/A
5. Pumps	2	I	B	B		A	N/A
6. Pump motors	*	I	N/A	B		A	1E
7. Valves, isolation LPCI line	1	I	A	B	(g)	D,A,C	1E
8. Valves, isolation other	2	I	B	B	(g)	D,C,A	1E
9. Pump suction from suppression pool, piping, valves, including isolation valves	2	I	B	B	(g)	A	1E
10. Valves, beyond isolation valves	2	I	B	B	(g)	A,C	1E
11. Mechanical modules	2	I	B	B	(g)	A	N/A
12. Electrical modules with safety function	*	I	N/A	B	(w)	N/A	1E
13. Cables with safety function	*	I	N/A	B		N/A	1E
14. Containment Spray piping and nozzles	2	I	B	B		C	N/A
15. Suction strainers	Other	I	D	B	(z)	C	N/A

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
XI. Low-Pressure Core Spray (Section 6.3) (v)							
1. Piping, within outboard isolation valve	1	I	A	B	(g)	A,C,D	N/A
2. Piping, beyond outboard isolation valve	2	I	B	B	(g)	A	N/A
3. Pumps	2	I	B	B		A	N/A
4. Pump motors	*	I	N/A	B		A	1E
5. Valves, isolation and within containment	1	I	A	B	(g)	A,D	1E
6. Valves, beyond outboard isolation valves	2	I	B	B	(g)	A	1E
7. Pump suction from suppression pool, piping, valves, including isolation valves	2	I	B	B	(g)	C,A	1E
8. Electrical modules with safety function	*	I	N/A	B	(w)	N/A	1E
9. Cables with safety function	*	I	N/A	B		N/A	1E
10. Suction strainer	Other	I	D	B	(z)	C	N/A
XII. High-Pressure Core Spray (Section 6.3) (v)							
1. Piping, within outboard isolation valve	1	I	A	B	(g)	C	N/A
2. Piping, return test line to RCIC storage tank beyond second isolation valve	Other	N/A	D	N/A		F,A	N/A
3. Piping beyond out-board isolation valve, other	2	I	B	B	(g)	F,A	N/A
4. Pump	2	I	B	B		F	N/A
5. Pump motor	*	I	N/A	B		F	1E
6. Valves, Outboard isolation and within containment	I	1	A	B	(g)	F,A	1E
7. Valves, beyond isolation valves, motor operated	2	I	B	B	(g)	F,A	1E
8. Valves, other	2	I	B	B	(g)	F,A	1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
9. Pump suction from suppression pool, piping, valves, including isolation valves	2	I	B	B	(g)	F,C,A	1E
10. Pump suction from RCIC storage tank, piping valves	2	I	B	B	(g)	M,F,A	1E
11. Electrical auxiliary equipment	*	I	N/A	B		X	1E
12. Electrical modules with safety function	*	I	N/A	B	(w)	N/A	1E
13. Cables with safety function	*	I	N/A	B		N/A	1E
14. Suction strainer	Other	I	D	B	(z)	C	N/A
XIII. <u>RCIC System</u> (Section 5.4.6) (v)							
1. Piping, within out-board isolation valves and up to the main steam	1	I	A	B	(g)	D,C,A	N/A
2. RCIC steam piping, beyond outboard isolation valve	2	I	B	B	(g)	A	N/A
3. RCIC turbine exhaust to suppression pool	2	I	B	B		A,C	N/A
4. RCIC pump suction from suppression pool, piping, valves, including isolation valves	2	I	B	B	(g)	A,C	1E
4a. RCIC Pump Suction from RCIC Storage Tank Piping & Valves	2	I	B	B	(g)	A,F,M	1E
5. RCIC pump discharge to RCIC storage tank within second isolation valve, piping, valves, including isolation valve	2	I	B	B	(g)	A,F,M	1E
6. RCIC pump discharge to RCIC storage tank beyond second isolation valve, piping, valves	Other	N/A	D	N/A		A,F,M	non-1E
7. Pump	2	I	B	B		A	N/A
8. Pump motors	*	I	N/A	B		A	1E
9. Valves, isolation and within	1	I	A	B	(g)	A,D	1E
10. Valves, other	2	I	B	B	(g)	A	1E
11. Turbine	2	I	N/A	B	(h)	A	N/A

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
12. Electrical modules with safety function	*	I	N/A	B	(w)	N/A	1E
13. Cables with safety function	*	I	N/A	B		N/A	1E
14. Suction strainer	Other	I	D	B	(z)	C	N/A
XIV. Fuel Service Equipment							
1. Fuel preparation machine	3	I	N/A	B		C,F	N/A
2. General purpose grapple	Other	N/A	N/A	B		C,F	N/A
XV. Reactor Vessel Service Equipment							
1. Steamline plugs	3	I	C	B		C	N/A
2. Dryer and separator strongback and RPV head strongback	Other	N/A	N/A	B		C	N/A
XVI. In-Vessel Service Equipment							
1. Control rod grapple	Other	N/A	N/A	B		C,A	N/A
XVII. Refueling Equipment							
1. Refueling equipment platform assembly	2	I	N/A	B		C	non-1E
2. Refueling bellows	Other	N/A	N/A	N/A		D	N/A
3. Bellows, blind flange and spool pieces that form a part of containment boundary	2	I	N/A	B		C	N/A
4. Fuel transfer tube	Other	I	D	N/A	(p)	C,F	N/A
5. Isolation valves (fuel transfer tube)	Other	N/A	D	N/A	(p)	C,F	N/A
6. Containment penetration sleeve, assembly (fuel transfer tube)	2	I	B	B	(p)	C,F	N/A
XVIII. Storage Equipment							
1. Fuel storage racks	2	I	N/A	B		C,F	N/A

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

PRINCIPAL COMPONENTS(s)		SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
2.	In-vessel rack (temporary)	2	I	N/A	B	(cc)	D	N/A
XIX. Radwaste System								
1.	Tanks, Atmospheric	Other	N/A	D	B	(g)(m)(q)	W	N/A
2.	Heat Exchangers	Other	N/A	D	B	(m)(q)	W	N/A
3.	Piping and valves forming part of containment boundary	2	I	B	B	(m)	C,F	1E
4.	Piping, other	Other	N/A	D	B	(g)(m)(q)	W	N/A
5.	Pumps	Other	N/A	D	B	(l)(m)	W	N/A
6.	Valves, flow control and filter system	Other	N/A	D	B	(l)(m)	W	non-1E
7.	Valves, other	Other	N/A	D	B	(g)(m)	W	non-1E
8.	Mechanical modules	Other	N/A	D	B	(m)	W	N/A
9.	Pump Motors	Other	N/A	N/A	N/A		W	non-1E
10.	Radwaste Packaging	Other	N/A	N/A	B,H	(gg)	W	N/A
XX. Reactor Water Cleanup System								
1.	Vessels: filter/demineralizer	Other	I	D	B		C	N/A
2.	Heat exchangers	Other	N/A	C	B		C	N/A
3.	Piping within drywell and drywell to containment penetration guard-pipe up to outboard containment isolation valve	1	I	A	B	(g)	A,C,D	N/A
4.	Piping from outboard con-tainment isolation valve to pumps, back to containment. Excluding penetration. All piping within containment. Blowdown piping downstream of outboard containment isolation valve up to shut-off valves	Other	I	C	B	(g)	A	N/A
5.	Pumps	Other	N/A	C	B		A,C	N/A
6.	Valves, drywell isolation valves	1	I	A	B	(g)	D	1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

PRINCIPAL COMPONENTS(s)		SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
7.	Valves within drywell, other than isolation valves	1	I	A	B	(g)	D	non-1E
8.	Valves, beyond outboard containment isolation valves	Other	N/A	D	N/A	(g)	A	non-1E
9.	Mechanical modules	Other	N/A	C	N/A		C	N/A
10.	Piping, return to reactor pressure vessel from inboard containment isolation valve up to junction with feedwater	2	I	B	B	(g)	A	N/A
11.	Piping, between inboard and outboard containment isolation valves	2	I	B	B	(g)	A,C,D	N/A
12.	Valves, containment isolation	2	I	B	B	(g)	C,A	1E
13.	Piping, downstream of blowdown shut-off valves to main condenser and radwaste	Other	N/A	D	N/A	(g)	A,T,W	N/A
14.	Filter/demineralizer precoat subsystem	Other	N/A	D	N/A		C	N/A
15.	Sample station	Other	N/A	D	N/A		C	N/A
16.	Nonregenerative heat exchanger shell and component cooling water piping	Other	N/A	D	N/A		C	N/A
17.	Pump Motors	*	N/A	N/A	N/A		A,C	non-1E
18.	Mitigation Monitoring System	N/A	N/A	D	N/A	(nn)	A	non-1E
XXI.	<u>Fuel Pool Cooling and Cleanup System</u>							
1.	Vessels, filter/demineralizers	Other	N/A	D	N/A	W	N/A	
2.	Vessels, other	3,Other	I,N/A	C	B,N/A		F	N/A
3.	Heat exchanges	3	I	C	B		F	N/A
4.	Piping	3,Other	I,N/A	C,D	B,N/A		F,C	N/A
5.	Pumps	3	I	C	B		F,C	N/A
6.	Valves and piping, containment isolation	2	I	B	B	(g)	C	1E
7.	Valves and piping filter demineralizers	Other	N/A	D	N/A		F,W	non-1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
8. Makeup system, shutdown service water system	3	I	C	B		A,C,F	N/A
9. Makeup to FPCC from condensate storage tanks	Other	N/A	D	N/A		F	non-1E
10. RHR connection	3	I	C	B	(u)	F	N/A
11. Fuel pool cooling pump motors	*	I	N/A	B	(qq)	F	1E
12. Cable with safety function	*	I	N/A	B		N/A	1E
13. Electrical modules with safety function	*	I	N/A	B	(w)	N/A	1E
XXII. Control Room Panels							
1. Electrical modules with safety function	*	I	N/A	B	(w)	N/A	1E
2. Cables with safety function	*	I	N/A	B		N/A	1E
XXIII. Local Panels and Racks							
1. Electrical modules with safety function	*	I	N/A	B		N/A	1E
2. Cables with safety function	*	I	N/A	B		N/A	1E
XXIV. Off-Gas System							
1. Tanks	Other	N/A	D	B	(q)(m)(aa)	T,A	N/A
2. Heat exchangers	Other	N/A	D	B	(q)(m)(aa)	T,A	N/A
3. Piping	Other	N/A	D	B	g,c,q)(m)(aa)	T,A	N/A
4. Pumps	Other	N/A	D	B	(c)(m)(aa)	T,A	N/A
5. Valves, flow control	Other	N/A	D	B	(c,q)(m)(aa)	T,A	non-1E
6. Valves, other	Other	N/A	D	B	(g)(m)(q)(aa)	T,A	non-1E
7. Pressure vessels	Other	N/A	D	B	(q)(m)(aa)	T,A	N/A
8. Pump motors	Other	N/A	N/A	N/A		T,A	non-1E
XXV. Shutdown Service Water Systems for Shutdown Equipment Cooling (Section 9.2.1.2) (v)							
1. Piping	3	I	C	B	(00)	A,N/F,O	N/A
2. Pumps	3	I	C	B		P	N/A

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
3. Pump motors	*	I	N/A	B	(jj)	P	1E
4. Piping and valves forming part of the containment boundary	2	I	B	B		C,A	1E
5. Valves, other	3	I	C	B	(00)	A,N/F,O	1E
6. Electrical modules with safety function	*	I	N/A	B	(w)	N/A	1E
7. Cables with safety-related function	*	I	N/A	B		N/A	1E
XXVI. <u>Plant Service Water Systems for Other Purposes</u>							
1. Piping and valves, other	Other	N/A	D	N/A		A,C,T	non-1E
2. Pumps	Other	N/A	D	N/A		P	N/A
3. Motors	Other	N/A	N/A	N/A		P	non-1E
XXVII. <u>Instrument, Breathing, and Service Air Systems</u>							
1. Vessels, accumulators supporting safety-related system	3	I	C	B		D,C	N/A
2. Piping and valves in lines between accumulators and safety-related systems	3	I	C	B		A,F,C	1E
3. Piping and valves forming part of containment boundary	2	I	B	B		A,C	1E
4. Control Room Emergency Breathing Air							
a) Piping, valves, instruments	Other	I	C	B		X	non-1E
b) Air bottles, filters, bottle piping	Other	N/A	D	N/A	(bb)	S,X	N/A
c) Pressure regulators	Other	I	D	B		X	N/A
5. Certain Instrument Air Regulators	3	I	N/A	B		S,X	N/A
6. ADS Accumulator Backup Air Supply							
a) Compressed Air Tank Farm and air filter (including Safety Relief Valves)	Other	I	D	B	(mm)	A	N/A

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
b) Remaining (except item 3) pipe/components between Compressed Air Tank Farms and ADS Accumulator	3	I	C	B		A,C,D	1E
7. Remaining air systems, non-safety related.	Other	N/A	D	N/A		A,C,D,F,N, S,T,W,X	non-1E
XXVIII. <u>Diesel Generator Systems</u>							
1. Fuel day tanks	3	I	C	B		S	N/A
2. Diesel fuel storage tanks	3	I	C	B		S	N/A
3. Piping and valves, fuel oil system							
a) off skid	3	I	C	B	(j)	S,O	1E
b) on skid	3	I	D	B	(ll)	S,O	1E
4. Pumps, fuel oil transfer system	3	I	C	B	(j)	S	N/A
5. Pump motors, fuel oil transfer system	*	I	N/A	B		S	1E
6. Starting Air System (Div. I and II)							
a) Air Start Skid							
1) Inlet Check Valve to Air Receiver Tanks to the Diesel	3	I	D	B	(kk)	S	N/A
2) Piping and Valves upstream of Air Receiver Tank Inlet Check Valve, downstream of Air Receiver Tank drain valve	Other	N/A	D	N/A		S	non-1E
3) Compressors	Other	I**	D	N/A		S	non-1E
** Non-safety compressors seismically supported.							
b) Engine Skid	3	I	D	B	(ll)	S	N/A
7. Starting Air System (Div. III)							
a) Air Start Skid							
1) Inlet Check Valve to Air Receiver Tanks to the Diesel	3	I	D	B	(kk)	S	N/A

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
2) Piping, Valves, Compressors upstream of Air Receiver Tank Inlet Check Valve, downstream of Air Receiver Tank drain valve	Other	N/A	D	N/A	(j)	S	non-1E
3) Compressors	Other	I**	D	N/A	(j)	S	non-1E
** Non-safety compressors seismically supported							
b) Engine Skid	3	I	D	B	(ll)	S	N/A
8. Diesel generators	2	I	N/A	B		S	1E
9. a) Diesel generator exhaust silencer and screen grating	Other	N/A	N/A	N/A			N/A
b) Diesel generator exhaust piping	3	I	D	B	(kk)	S	N/A
c) Exhaust piping on diesel engine	3	I	D	B	(ll)	S	N/A
10. Electrical modules with safety function	*	I	N/A	B	(w)	N/A	1E
11. Cables with safety function	*	I	N/A	B		N/A	1E
12. a) Diesel Engine Cooling Heat Exchanger (ASME Section III, Code 'N' Stamped)	3	I	C	B		S	1E
b) Diesel Engine Cooling Piping (ASME Section III Design Only) and Engine Mounted Piping and Cooling Jacket (DEMA Standards)	3	I	C	B		S	1E
13. Diesel engine lubrication system	3	I	D	B		S	1E
14. Combustion air system to filter inlet	3	I	D	B		S	N/A
15. Air dryers and associated piping, controls, and cables	Other	I	D	N/A		S	N/A
XXIX Standby Gas Treatment System							
1. Filter Units	2	I	N/A	B		A	N/A
2. Air piping	2	I	N/A	B		M	N/A
3. Cables with safety functions	*	I	N/A	B		N/A	1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
4. Fans/dampers	2	I	N/A	B		X	1E
5. Electrical heaters, instrumentation and controls with safety function	2	I	N/A	B	(w)	N/A	1E
XXX. Power Conversion System							
1. Main steam piping from second isolation valve to & incl. first shutoff valve	2	I	B	B		A	N/A
2. Main steam branch piping between the second isolation valve and the first shutoff valve, from branch point at main steam piping to and including the first valve in the branch line	2	I	B	B		A	N/A
3. Main steam piping between the shutoff valve and the turbine main stop valve	Other	N/A	D	N/A		A,T	N/A
4. Turbine bypass piping	Other	N/A	D	N/A		T	N/A
5. Main steam branch piping between the first shutoff valve downstream of the second isolation valve and the turbine main stop valve.	Other	N/A	D	N/A		A,T	N/A
6. Turbine stop valves, turbine control valves and turbine bypass valves.	Other	N/A	D	N/A	(i)	T	non-1E
7. Main steam leads from turbine control valves to turbine casing	Other	N/A	D	N/A		T	N/A
8. Feedwater and condensate system beyond 3rd isolation valve	Other	N/A	D	N/A	(d)	A,T	non-1E
9. Turbine Generator	Other	N/A	N/A	N/A		T	non-1E
10. Valves, instrumentation beyond outermost isolation valves	Other	N/A	D	N/A	(g)	A,T	non-1E
11. Condenser	Other	N/A	N/A	N/A		T	N/A
12. Air ejection equipment	Other	N/A	D	N/A		T	non-1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
13. Feedwater treatment system	Other	N/A	D	N/A		T	non-1E
14. Turbine bypass system beyond turbine bypass valve	Other	N/A	D	N/A		T	non-1E
15. Turbine gland sealing system components	Other	N/A	D	N/A		T	non-1E
16. Heater drain piping	Other	N/A	D	N/A		T	N/A
17. Heater drain valves	Other	N/A	D	N/A		T	non-1E
XXXI. <u>ECCS Equipment Area Cooling</u>							
1. Fans and ductwork with a safety function	3	I	N/A	B	(n)	A	1E
2. Instrumentation and control with safety function	3	I	N/A	B		A	1E
3. Electrical modules and cables with safety function	*	I	N/A	B	(w)	F,A	1E
4. Heat exchanger and pressure retaining components and valves	3	I	C	B	(n)	A,F	N/A
XXXII. <u>Condensate Storage and Transfer</u>							
1. RCIC condensate storage tanks	Other	N/A	D	N/A		O	N/A
2. Condensate storage tanks	Other	N/A	D	N/A		O	N/A
3. Piping and valves, containment penetration	2	I	B	B		O	1E
4. Piping, valves, other	Other	N/A	D	N/A		O	non-1E
XXXIII. <u>Auxiliary A-C Power System</u>							
1. All equipment necessary for operation of Safety Class 2 Mechanical Systems	*	I	N/A	B		N/A	1E
2. All equipment necessary for operation of Safety Class 3 Mechanical Systems	*	I	N/A	B		N/A	1E
3. Other Auxiliary A-C Power Equipment	Other	N/A	N/A	N/A		M	non-1E
4. Containment Electrical Penetration Assemblies	*	I	N/A	B	(ii)	C	1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

PRINCIPAL COMPONENTS(s)		SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
XXXIV.	Miscellaneous Components							
1.	Containment Polar Crane	3	I	N/A	B		C	non-1E
2.	Fuel Building Crane	3	I	N/A	B		F	non-1E
3.	Turbine Building Cranes	Other	N/A	N/A	N/A		T	non-1E
XXXV.	Civil Structures (Section 3.8)							
1.	Containment	N/A	I	N/A	B			N/A
2.	CGCB structure	N/A	I	N/A	B	(c)		N/A
3.	Auxiliary building	N/A	I	N/A	B			N/A
4.	Fuel building	N/A	I	N/A	B			N/A
5.	Control building	N/A	I	N/A	B			N/A
6.	Diesel generator and HVAC building	N/A	I	N/A	B			N/A
7.	Radwaste substructure	N/A	I	N/A	B			N/A
8.	Circulating water screen house	N/A	I	N/A	B			N/A
9.	Turbine building	N/A	N/A	N/A	N/A			N/A
10.	Service building	N/A	N/A	N/A	N/A			N/A
11.	Ultimate heat sink	N/A	I	N/A	B			N/A
12.	Radwaste building, above grade	N/A	N/A	N/A	N/A			N/A
13.	RPV Pedestal	N/A	I	N/A	B			N/A
14.	Safety-related masonry walls	N/A	I	N/A	B		A,C,E,F,S,W,X, H	N/A
15.	Fuel pools and pool liners	N/A	I	N/A	B		F	N/A
XXXVI.	Other Structures							
1.	Those supporting or protecting safety related equipment	N/A	I	N/A	B			N/A
2.	Cable trays and tray hangers, conduit and conduit hangers in Seismic Category I areas.	N/A	I	N/A	B	(hh)		N/A
XXXVII.	Miscellaneous							
1.	Component cooling water system	2/3/Other	I,N/A	B/C/D	B,N/A		A,F,C,D,M,X	1E, non-1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

	PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
	a) Valves and piping containment isolation	2	I	B	B		A,C,D	1E
	b) Valves and piping for supplying shut-down service water to the fuel pool heat exchangers	3	I	C	B		F	1E
	c) Valves and piping for supplying shutdown service water to the Recirculation Pump Seals	Other	I	C	B		D	1E
	d) Piping and valves	Other	N/A	D	N/A		A,F,C,D	N/A
	e) Heat exchangers	Other	N/A	D	N/A		M,X	N/A
	f) Storage tank	Other	N/A	D	N/A		M,X	N/A
	g) Instrumentation and controls with safety function	2/3	I	N/A	B		A,C,D	1E
2.	Turbine building, closed cooling water system	Other	N/A	D	N/A		T,W	non-1E
3.	Shutdown service water equipment area cooling							
	a) Fans, ductwork, valves, with safety function	3	I	N/A	B		B	1E
	b) Heat exchanger Pressure-retaining components and valves	3	I	C	B		H	N/A
	c) Instrumentation and controls with safety functions	3	I	N/A	B	(w)	H	1E
	d) Electrical modules and cables with safety function	*	I	N/A	B	(w)	H	1E
4.	Switchgear heat removal							
	a) Fans, ductwork, dampers valves, with safety function	3	I	N/A	B		A,X	1E
	b) Heat exchanger pressure-retaining components and valves	3	I	C	B		A,X	N/A
	c) Instrumentation and control with safety function	3	I	N/A	B	(w)	A,X	1E
	d) Electrical modules and cables with safety function	*	I	N/A	B	(w)	A,X	1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
e) Refrigeration piping with safety function	Other	I	N/A	B		A,X	N/A
5. Control room and Technical Support Center HVAC							
a) Fans, fan motors, dampers, ductwork	3	I	N/A	B		X	1E
b) Heat exchanger pressure-retaining components, valves and pumps	3	I	C	B		X	N/A
c) Electric heater instru-mentation and controls with safety function	3	I	N/A	B	(w)	S	1E
d) Electrical modules and cables with safety function	*	I	N/A	B	(w)	X	1E
e) Filter units	3	I	N/A	B		X	1E
6. Diesel generator HVAC							
a) Fan and ductwork with safety function	3	I	N/A	B		S	1E
b) Instrumentation and control with safety function	3	I	N/A	B	(w)	S	1E
c) Electrical modules and cables with safety function	*	I	N/A	B	(w)	S	1E
7. Combustible gas control system cooling							
a) Fan and ductwork with safety function	3	I	N/A	B	(rr)	C	1E
b) Heat exchanger pressure-retaining components and valves	3	I	C	B	(rr)	C	N/A
c) Instrumentation and controls with safety function	3	I	N/A	B	(w) (rr)	C	1E
d) Electrical modules and cables with safety function	*	I	N/A	B	(w) (rr)	C	1E
8. Fire protection system	Other	N/A	N/A	B	(r)		non-1E
9. Standby gas treatment system and hydrogen recombiner equipment area cooling							
a) Fan and ductwork with safety function	3	I	N/A	B		C	1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
b) Heat exchanger pressure-retaining components	3	I	C	B		C	N/A
c) Instrumentation and controls with safety function	3	I	N/A	B		C	1E
d) Electrical modules and cable with safety function	*	I	N/A	B		C	1E
10. MSIV leakage room cooling							
a) Fan and ductwork with safety function	3	I	N/A	B		A	1E
b) Heat exchanger pressure-retaining components and valves	3	I	C	B		A	N/A
c) Instrumentation and controls with safety function	3	I	N/A	B		A	1E
d) Electrical modules and cable with safety function	*	I	N/A	B		A	1E
11. Meteorological data collection equipment	Other	N/A	N/A	N/A		O,X	non-1E
12. Postaccident sample system							
a) Sample analysis panel	Other	N/A	D	N/A	(bb)	S	non-1E
b) Sample monitor panel	Other	N/A	N/A	N/A	(bb)	S	non-1E
c) Closed loop cooling system	N/A	N/A	D	N/A	(bb)	S	non-1E
d) Containment isolation valves	2	I	B	B		A,C	1E
e) Sample block and back flush valves	Other	N/A	D	N/A		A	non-1E
f) Drywell floor and equipment drain sump sample pumps	Other	N/A	D	N/A	(bb)	D	non-1E
g) Containment floor and equipment drain sump sample pump	Other	N/A	D	N/A	(bb)	S	non-1E
13. Postaccident monitoring systems							
a) Piping, within containment pressure boundary and/or with post-LOCA function	2,3	I	N/A	B		C	N/A
b) Valves, within containment pressure boundary and/or with post-LOCA function	2	I	N/A	B		C	1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
c) Electrical and instrumentation modules with post-LOCA function	2	I	N/A	B		C,X	1E
d) Cables with safety function	2	I	N/A	B		C,A,X	1E
14. Equipment for emergency response facilities							
a) Airborne and Area Radiation Monitors for TSC	Other	N/A	N/A	N/A	(y)	M	non-1E
b) Electric Modules with display function for TSC	Other	N/A	N/A	N/A	(y)	M	non-1E
15. Fuel Bldg. HVAC system dampers and ductwork with safety function	3	I	N/A	B	(w)	F.S	1E
16. Hydrogen Water Chemistry (HWC) System	N/A	N/A	D	N/A		T, W, O	non-1E
XXXVIII. <u>Combustible Gas Control System</u>							
1. Hydrogen Recombiner System	2	I	B	B	(rr)	C,A,X	1E
2. Standby Gas Treatment System (See Item XXIX)							
3. Containment/Drywell Mixing System	2	I	B	B		D,C	1E
4. Containment/Drywell Monitoring Systems	2,3, Other	I	B, C, D	B		D,C	1E
5. Electrical modules with safety function	*	I	N/A	B	(w)	N/A	1E
6. Vacuum relief system	2	I	B	B		C,D	1E
XXXIX. <u>Suppression Pool Make-up System (SM)</u> (Section 6.2.7) (v)							
1. Piping	2	I	B	B		C	N/A
2. Valves with safety function	2	I	B	B		C	1E
3. Electrical modules with safety function	*	I	N/A	B	(w)	N/A	1E
4.						C	
Other valves	Other	I	B	B			non-1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

PRINCIPAL COMPONENTS(s)	SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI-FICATION(d)	QUALITY ASSURANCE REQUIRE-MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
XL. <u>Auxiliary D-C Power System</u>							
1. Divisional batteries, chargers and distribution panels	*	I	N/A	B		A	1E
2. Non-divisional batteries, chargers and distribution panels	Other	N/A	N/A	N/A		X	non-1E
XLI. <u>Suppression Pool Clean-up System (SF)</u>							
1. Piping and valves forming part of Containment Boundary	2	I	B	B		C,A	1E
2. Other piping and valves	Other	N/A	D	N/A		C,A,T,W	non-1E
XLII. <u>Demineralizer makeup system</u>							
1. Piping and valves	Other	N/A	D	N/A		P	non-1E
2. Tanks	Other	N/A	D	N/A		O	N/A
3. Filters	Other	N/A	D	N/A		D	N/A
4. Pumps and motors	Other	N/A	D	N/A		P	non-1E
5. Pretreated filtration components	Other	N/A	D	N/A		W	non-1E
6. Demineralizing train components	Other	N/A	D	N/A		P	non-1E
XLIII. <u>Main steamline isolation valve leakage control system</u> (pp)							
1. Piping and valves up to the first isolation vale of the inboard subsystem	1	I	A	B		A	1E
2. Piping and valves, other	2	I	B	B		A	1E
3. Blowers/motors	*	I	N/A	B		A	1E
4. Heaters	Other	I	N/A	B		A	1E
5. Electric modules with safety function	*	I	N/A	B	(w)	N/A	1E
6. Cables with safety function	*	I	N/A	B		N/A	1E
XLIV. <u>Containment Isolation</u>							
1. Piping between the inside and	2	I	B	B		A,C,D	1E

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (CONTINUED)

PRINCIPAL COMPONENTS(s)		SAFETY CLASS(b)	SEISMIC CATEGORY(c)	QUALITY GROUP CLASSI- FICATION(d)	QUALITY ASSURANCE REQUIRE- MENTS(e)	COMMENTS	LOCATION(s)	ELECTRICAL CLASSIFICATION(t)
	outside containment isolation valves including the isolation valves for systems with no safety function which penetrate the containment							
2.	Containment penetration assemblies, except drywell guardpipe bellows	2	I	B	B		A,C,D	1E
3.	Drywell guardpipe bellows	Other	I	D	B	(ff)	C	N/A

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

NOTES

(a) The word "within" as defined in the table defines those portions of systems, components or equipment in the piping direction from the component toward the Reactor Pressure Vessel regardless of whether a portion of the system is influent or effluent. The word "beyond" defines those portions of system, components or equipment in the piping direction from the component away from the Reactor Pressure Vessel.

A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electrical modules include sensors, power supplies, and signal processors; mechanical modules include turbines, strainers and orifices.

(b) 1, 2, 3, or Other are safety classes defined in Subsection 3.2.3. N/A and * are also denoted in subsection 3.2.3.

(c) I = The equipment is constructed in accordance with the seismic requirements of Seismic Category I structures and equipment as described in Section 3.7.

All civil structures classified as Seismic Category I are designed for the effects of CPS natural phenomena such as tornado, wind loads, external missiles, floods, etc., except the containment gas control boundary building (CGCB). The CGCB is a Seismic Category I structure capable of withstanding all of CPS natural phenomena except the tornado and external missiles.

N/A= The seismic requirements for the safe shutdown earthquake are not applicable to the equipment.

(d) A, B, C, D, - NRC quality groups defined in Regulatory Guide 1.26. The equipment is constructed in accordance with the codes listed in Table 3.2-3.

N/A = Quality Group Classification not applicable to this equipment. All piping systems penetrating and forming a part of the containment barrier or structure are at least Quality Group B, Quality Assurance B, Seismic Category I at the penetration location and up to and including the containment isolation valves. Beyond isolation valves, classification can change to that detailed in the table.

(e) B = Quality Assurance program described in Chapter 17 implements the requirements of 10 CFR 50, Appendix B and is applicable to this equipment. H = Quality Assurance Program described in the Clinton Power Station Quality Assurance Manual implements the requirements of 10CFR71, Subpart H and is applicable to this process.

N/A = Quality Assurance program requirements are not applicable to this equipment.

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

(f) The control rod drive insert and withdrawal lines from the drive flange up to and including the first valve on the hydraulic control unit is Safety Class 2.

(g) 1. Lines 3/4 inch and smaller which are part of the reactor coolant pressure boundary are Safety Class 2.2. Instrument sensing lines are classified as follows:

- <u>For Quality Group A Process Lines</u>:

  The branch line (for 3/4 inch and smaller) to and including the root valve shall be Quality Group B. The instrument sensing line (downstream of the root valve) shall be Quality Group B for pipe and Quality Group S for tubing.

- <u>For Quality Group B Process Lines</u>:

  The branch line to and including the root valve shall be Quality Group B. The instrument sensing line (downstream of the root valve) shall be Quality Group C for pipe and Quality Group T for tubing.

- <u>For Quality Group C Process Lines</u>:

  The branch line to and including the root valve shall be Quality Group C. The instrument sensing line (downstream of the root valve) shall be Quality Group C for pipe and Quality Group T for tubing.

- <u>For Quality Group D Process Lines</u>:

  The branch line to and including the root valve shall be Quality Group D. The instrument sensing line (downstream of the root valve) shall be Quality Group D for pipe and tubing.

- <u>For Quality Group A, B, or C Process Lines</u>:

  For instrument sensing lines that are used only during testing and are isolated by at least one normally-closed root valve, the instrument line (downstream of the root valve) shall be Quality Group D for pipe and tubing.

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

(h) The RCIC turbine is categorized as machinery and thus does not fall within the quality classification groups as earlier identified. To assure that the turbine is fabricated to the standards commensurate with performance requirements, General Electric has established specific requirements for this component which are as follows:

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

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

2. All pressure-containing castings and fabrications were hydrotested in 1.5 x design pressure.

3. All high-pressure castings were radiographed according to:

   ASTM E-94

E-142	Maximum feasible volume
E-71, 186 or 280	Severity level 3

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

5. Wheel and shaft forgings were ultrasonically tested according to ASTM A-388.

6. Butt-welds were radiographed according to ASME, and magnetic particle or liquid penetrant tested according to ASME Boiler and Pressure Vessel Code. Acceptance standards were in accordance with ASME Section III Paragraph NB-5340, NC-5340, NB-5350, NC-5350, respectively.

7. Notification was made on major repairs and records maintained thereof.

8. Record system and traceability is according to ASME Section III, NA-4000.

9. Control and identification is according to ASME Section III, NA-4000.

10. Procedures conformed to ASME Section III, NB-5100, NC-5100.

11. Inspection personnel were qualified according to ASME Section III, NB-5500, NC-5500.

(i) A certification was obtained from the manufacturers of the turbine stop valves and turbine bypass valves that all cast pressure-retaining parts of a size and configuration for which volumetric examination methods are effective have been examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards was permitted as an alternate to radiographic methods.

Examination procedures and acceptance standards were at least equivalent to those specified as supplementary types of examination in ANSI Std B31.1 Code, Paragraph 136.4.3.

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

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

(j) Those portions of piping and valves supplied by the diesel generator manufacturer meet the requirements of Safety Class 3, Quality Group C, Seismic Category I, although an "N" stamp will not be provided.

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

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

The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example, (1) all welds are LP inspected, (2) all socket welds are inspected for gap between pipe and socket bottom, (3) all welding is performed by qualified welders, and (4) all work is done per written procedures.

Code Group D is generally applicable because the codes and standards invoked by that group contain clauses that permit the use of manufacturer's standards and proven design techniques that are not explicitly defined within the codes of Code Group A, B, or C. This is supplemented by the QC techniques described above.

(l) ASME Section VIII and ANSI B31.1.0 apply downstream of outermost isolation valves.

(m) Only those portions of the Quality Assurance program described in Section 17.2 which meet 10 CFR 50 Appendix B and as delineated in Regulatory Guide 1.143, will be applied to this system as appropriate.

The radwaste systems piping, pumps and valves containing radwaste were designed and constructed in accordance with the applicable codes of Quality Group D and the requirements of Regulatory Guide 1.143 (formerly BTP 11-1). Quality Assurance program requirements applied during this phase were independent of the Quality Assurance program described in Section 17.1.

The scope of pressure testing includes all pressure-retaining components, appurtenances, and completed systems. Bolts, studs, nuts, washers, gaskets, and possible localized instances of pump and valve packing (e.g., packing leaking) are exempted from the pressure test. This is consistent with ASME Section III and ANSI B31.1 (1983 edition). Non-pressure retaining attachments and appurtenances to solid

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

waste tanks are exempted from the above requirements. Examples of this type of equipment are nozzles, piping and welds in atmospheric tanks which do not carry process fluid and whose elevation is above the tank overflow level. See USAR Section 1.8, Reg Guide 1.143, item number 8.

For radwaste system piping and components, the following quality control procedures of the contractors or subcontractors were required to be submitted to the AE for review:

1) Welding Procedures and Welding Procedure Qualifications

2) Hydrostatic/Pneumatic Tests

3) Nondestructive Examination (where applicable)

The following documentation was required to be submitted to the owner for review and acceptability:

1) Manufacturer's Certificates of Compliance or Certified Material Test Reports were required for pressure-retaining components.

2) Hydrostatic/Pneumatic Test Reports

3) Nondestructive Examinations Reports (where applicable)

(n) All pressure-retaining components are Quality Group C, Quality Assurance B, and Seismic Category I.

(o) Only equipment associated with a safety action (e.g. isolation) need conform to safety requirements.

(p) The portion of the transfer tube and the parts immediately adjacent and attached to that portion used for containment isolation are Quality Group B. The remainder of the tube and the valves are Quality Group D. For those parts of the fuel transfer system, such as the sheave bore, fuel carrier up-ender, and similar parts for which no codes exist, quality group clarification did not apply.

(q) Screwed connections backed up by seal welding or mechanical joints are permitted only on lines greater than 3/4-inch nominal pipe size and under 2-1/2 inches.

(r) Only those portions of the Quality Assurance program described in Chapter 17 which meet 10 CFR 50 Appendix B and as delineated in Regulatory Guide 1.120 will be applied to this system, as appropriate.

Quality assurance requirements for fire protection systems are applicable to those fire detection, suppression and extinguishing systems and components serving the following safety related structures or buildings; this includes connecting piping, wiring, or equipment that may be routed through or located within other areas, but which serve the following:

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

1. containment building,

2. control building,

3. diesel-generator building,

4. fuel building,

5. auxiliary building, and

6. screen house.

Additionally, quality assurance requirements for fire protection systems are applicable to the underground water main loop, including any branch connections, up to and including the first isolation valve outside of non-safety-related buildings or structures.

(s) A = auxiliary building

C = part of, or within, containment

D = drywell

E = within containment gas control boundary

F = fuel building

L = offsite

M = any other location

O = outdoors onsite

P = pump house

S = diesel generator building

T = turbine building

W = radwaste building

X = control building

H = circulating water screen house

N/A = not applicable

(t) Electrical Classification

Structures, systems and components shall be classified as "Class 1E", "Non-Class 1E", or "N/A" (Not Applicable) in accordance with the following:

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

1) Class 1E

The safety classification of the electric equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment.

2) Non-Class 1E

"Non-Class 1E" is the classification to be applied to all electric structures, systems and components other than "Class 1E" structures, systems and components.

3) N/A (Not Applicable) Classification

"N/A" indicates that the electrical classification is "not applicable" to this structure, system or component. For example, an assembly (such as a water strainer) may have no electrical components.

(u) To comply with Regulatory Guides 1.26 and 1.29, the RHR and SX systems are interconnected to the fuel pool, thereby providing redundant Seismic Category I sources of coolant to the fuel pool. Additionally, systems for maintaining water quality and quantity are designed so that any malfunction or a failure in such systems will not cause significant loss of inventory.

(v) This is a safety-related structure, system, or component located inside containment required for safe shutdown.

(w) Includes instruments and controls as detailed in Chapter 7.

(x) Subsystems required for postaccident monitoring include containment pressure monitoring, containment high-range radiation monitoring, and suppression pool water level monitoring.

(y) While this equipment is not safety-related, its operability and calibration will be controlled by the applicable requirements of the operational quality assurance program described in Section 17.2.

(z) The ECCS and RCIC suction strainer is designed so that stress is less than ASME B&PV Code Section III allowables, and welding and NDE use code requirements as guidelines.

(aa) Quality Assurance Program requirements are not applicable to the glycol subsystem of the off-gas system, except for the cooler condensers and regenerator chillers.

(bb) While this equipment is not safety related, it's operability and calibration will be controlled by Plant Administrative Procedures, preventive Maintenance Program, design change control procedures, and applicable portions of the operational Quality Assurance

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

Program in section 17.2 covering document control, corrective action, quality assurance records and audits to assure its operability during post accident periods.

(cc) Contingency tool, used only when RPV is open and temporary storage required within RPV. Not in place during plant operations.

(dd) Modifications to reactor internals including verification of structural integrity are subject to applicable portions of the CPS design control program. Administrative controls are in place to verify that reactor internals are properly installed following work activities that require their removal.

(ee) Scram solenoid valves are Class 1E components. The other electrical components are not safety-related and therefore non-1E components. Their failure will not affect the safety function of the HCU.

(ff) The drywell guardpipe bellows meet the requirements of ANSI B31.1, but are safety-related, Seismic Category I components.

(gg) The quality assurance requirements of 10CFR71, Subpart H have been incorporated into the Clinton Power Station Quality Assurance Program. This program was originally based solely on 10CFR50, Appendix B.

(hh) Quality assurance program requirements are not applicable to lighting and communications conduits/conduit supports or non-safety, non-seismic instruments mounted in Seismic Category I structures.

(ii) Containment electrical penetration assemblies servicing safety-related equipment are Class 1E. Others are non-Class 1E.

(jj) Pump motor bearing cooler was supplied as an integral part of the Division 1 and Division 2 motors and is subsequently seismically qualified with the motor per IEEE 344.

(kk) These portions of the piping and valves are designed, manufactured and inspected to ANSI B31.1 requirements and analyzed to ASME Section III allowables.

(ll) The engine mounted piping and components, from engine block to the engine interface are considered part of the engine assembly. This piping and associated components such as valves, fabricated headers and special fittings, are designed, manufactured and inspected in accordance to the requirement of the Diesel Engine Manufacturer Association (DEMA) Standards.

(mm) These components were originally purchased and installed as Non-Q. Evaluation and analysis have been done which qualified the subject components for seismic, EQ and operability (as applicable). Future maintenance and operability will be controlled by plant administrative procedures, Preventive Maintenance Program, design change control procedures, and applicable portions of the operational Quality Assurance Program referenced in Section 17.2. This program covers document control, corrective action, quality assurance records and audits to assure operability during post accident periods.

TABLE 3.2-1
CLASSIFICATION OF SYSTEMS, COMPONENTS AND STRUCTURES (continued)

(nn) The Mitigation Monitoring System is labeled with a “HX” designator as it is associated with the Hydrogen Water Chemistry system. Because the Mitigation Monitoring System is physically connected to the Reactor Water Cleanup System, it is listed under that system.

(oo) The sample cooler of Standby Gas Treatment System (SGTS) Exhaust, associated small bore piping and components are not classified as Quality Group C. However that portion is designed to Quality Group C standards.

(pp) As a result of the re-analysis of the Loss of Coolant Accident (LOCA) using Alternative Source Term (AST) Methodology, it is no longer necessary to credit the Main Steam Isolation Valve Leakage Control System (MSIVLCS) for post-LOCA activity leakage mitigation. The system has been left in place as a passive system and is not required to perform any safety function.

(qq) The Fuel Pool Cooling pump motors have air-to-water heat exchangers of Safety Class 3 and Quality Group C.

(rr) Due to revision of 10 CFR 50.44 the Combustible Gas System is no longer a safety related system and will be used during an accident that resulted in a degraded core (beyond design basis). This information is historical and kept here to understand the original design and construction of the system.

TABLE 3.2-2
SUMMARY OF SAFETY CLASS DESIGN REQUIREMENTS (MINIMUM)

	Safety Class			
Design Requirements	1[(d)]	2	3	Other
Quality Group Classification (a)	A	A,B,S,D,N/A	C,T,D,N/A	B,S,C,T,D,N/A
Quality Assurance Requirement (b)	B	B	B	B, N/A
Seismic Category (c)	I	I	I, N/A	I, N/A

Notes:

(a) The equipment has been constructed in accordance with the indicated code group listed in Table 3.2-1 and defined in Table 3.2-3.

(b) B - The equipment has been constructed in accordance with the quality assurance requirements of 10 CFR 50, Appendix B, as delineated in Chapter 17.

N/A - A quality assurance program is not applicable to this equipment.

(c) I - The equipment of these safety classes has been constructed in accordance with the seismic requirements for the safe-shutdown earthquake as described in Section 3.7.

N/A - The seismic requirements for the safe-shutdown earthquake are not applicable to the equipment of this classification.

(d) Application of code edition and addenda for the Reactor Coolant Pressure Boundary Class 1 Equipment is provided in Table 3.2-4.

TABLE 3.2-3
CODE CLASSIFICATION GROUPS - INDUSTRY CODES AND STANDARDS
FOR MECHANICAL COMPONENTS (a, b)*
ASME SECTION III CODE APPLICABLE SUBSECTIONS

QUALITY GROUP CLASSIFICATION	ASME SECTION III CODE CLASSES	PRESSURE VESSELS AND HEAT EXCHANGERS	PUMPS, VALVES, AND PIPING	METAL CONTAINMENT COMPONENTS	STORAGE TANKS (0-15 psig)	STORAGE TANKS ATMOSPHERE
A	1	NA & NB TEMA C	NA & NB Note (c)	---	---	---
B,S Note (g) (i)	2 MC	NA & NC TEMA C	NA & NC Note (c)	NA & NE	NA & NC	NA & NC
C,T Note (h) (i)	3	NA & ND TEMA C	NA & ND Note (c)	---	NA & ND	NA & ND
D Note (j)		ASME Sect. VIII Div.1	Piping and Valves B31.1 Note (f)	---	API-620 or equivalent Note (e)	API-650 AWWA-D100 ANSI B96.1 or equivalent Note (e)
		TEMA C	Pumps Note (d)			

*See following page for alphabetic notes.

TABLE 3.2-3 (Cont'd)

NOTES:

(a) With options and additions as necessary for service conditions and environmental requirements.

(b) Components of the reactor coolant pressure boundary meet the requirements of 10 CFR Part 50, Section 50.55a, "Codes and Standards", and its amendments of the NRC regulations to meet the codes and addenda in effect based on issuance of the construction permit.

(c) For pumps classified A, B, or C, applicable subsections NB, NC or ND, respectively, in ASME Section III Boiler and Pressure Vessel Code have been used as a guide in calculating the thickness of pressure-retaining portions of the pump and in sizing cover bolting.

(d) For pumps classified in Group D, ASME Section VIII Div. 1 has been used as a guide in calculating the wall thickness for pressure-retaining parts and in sizing the cover bolting.

(e) Tanks shall be designed to meet the intent of API, AWWA, and/or ANSI 96.1 Standards as applicable.

(f) For piping and valves classified in Group D that may be hydrostatically tested, the scope of pressure testing includes all pressure retaining components, appurtenances, and completed systems. Bolts, studs, washers, gaskets, and possible localized instances of pump and valve packing are exempted from the pressure test. This is consistent with ANSI B31.1 (1983 edition) and ASME Section III NB6000.

(g) Quality Group S nuclear safety-related instrumentation tubing is designed, fabricated, installed, examined and tested in accordance with the ASME Code Section III, Subsection NC requirements, except that no authorized inspector involvement or N-stamp is required.

(h) Quality Group T nuclear safety related instrumentation tubing is designed, fabricated, installed, examined and tested in accordance with the ASME Code Section III, Subsection ND requirements, except that no authorized inspector involvement or N-stamp is required.

(i) The use of revised allowable stress values within the 2001 Edition through 2003 Addenda has been reconciled and is acceptable for design evaluations, modifications, repairs, and replacements for ASME Class 2 and 3 piping and components.

(j) The use of revised allowable stress values within the 2004 Edition through 2005 Addenda has been reconciled and is acceptable for design evaluations, modifications, repairs, and replacemtns for B31.1 piping and components.

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TABLE 3.2-4
QUALITY GROUP A COMPONENTS-APPLICABLE ASME CODE
EDITION AND ADDENDA

NSSS Components

Component Description	Master Parts List Number	ASME Section III Code Edition and Addenda
1. Reactor recirculation system		
a. Gate valves	B33-F023 B33-F067	1971 Edition, Winter 1973 Addenda
b. Flow control valves	B33-F060	1974 Edition, Summer 1976 Addenda
1. body		
2. topworks		
c. Pump	B33-C001	1971 Edition, Winter 1973 Addenda
d. Piping	B33-G001	1974 Edition, Summer 1974 Addenda
2. Main steam system		
a. Main steam isolation valves	B21-F022 and B21-F028	1974 Edition, Addenda is not applicable
b. Safety/relief valves	B21-F041 B21-F047 B21-F051	1974 Edition, Summer 1976 Addenda
c. Piping	B21-G001	1974 Edition 1975 Edition, Summer
3. High-pressure core spray system		
a. Valves	E22-F004	1971 Edition, Winter 1973 Addenda
4. Reactor Pressure Vessel	B13-D003	1971 Edition, Summer 1973 Addenda

TABLE 3.2-4
QUALITY GROUP A COMPONENTS APPLICABLE ASME CODE
EDITION AND ADDENDA (Continued)

NON-NSSS Components

(Except the control rod drive system components that are designed, fabricated and erected by Reactor Controls, Inc.)

Component Description	ASME Section III Code Edition and Addenda
Piping (all sizes)	1974 Edition, Summer 1974 Addenda
Hangers, snubbers, and supports*	1974 Edition, Summer 1974 Addenda
Valves - 2 1/2 inches and larger (all types)	1974 Edition, Summer 1975 Addenda
Valve - 2 inches and smaller (all types)	1974 Edition, Winter 1975 Addenda

*Except for AISC Supplementary Structural Steel members/items, refer to Subsection 3.9.3.4.2.1.

NON-NSSS Components (Quality Group B)

(Control rod drive system components designed, fabricated and erected by Reactor Controls, Inc.)

Component Description	ASME Section III Code Edition and Addenda**
Piping (all sizes)	1977 Edition, Summer 1978 Addenda
Hangers, snubbers, and supports	1977 Edition, Summer 1978 Addenda

** Note: The use of revised allowable stress values for ASME Section III, Class 2 and 3 piping components within the 2001 Edition through 2003 Addenda has been reconciled and is acceptable for design evaluations, modifications, repairs, and replacemetns.

3.3 WIND AND TORNADO LOADINGS

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

A design wind velocity of 85 mph is used in the design of Clinton Power Station (CPS) Seismic Category I structures. This wind velocity is based on a mean recurrence interval of 100 years using the wind speed distribution map given in ANSI A58.1-1972 (See Figure 3.3-1).

The effective velocity pressures of different heights are obtained using Table 5 for exposure C of ANSI A58.1-1972. The effective velocity pressures include a variation of wind velocity given by the following formula:

$$V_z = V_{30}\left(z/30\right)^{1/7} \qquad (3.3\text{-}1)$$

where

V_{30} = design wind velocity in mph at height of 30 feet above grade.

V_z = wind velocity in mph at a height of z feet above grade.

The effective velocity pressures also include the effect of gusts through the use of appropriate gust factors as specified in ANSI A58.1-1972.

3.3.1.2 Determination of Applied Forces

The design wind velocity V is converted to velocity pressure using the formula:

$$q_{30} = 0.00256\ V^2{}_{30} \qquad (3.3\text{-}2)$$

where

q_{30} = basic wind pressure in psf.

The effective velocity pressures of winds for buildings and structures, q , and for parts and portions, q , at various heights above the ground are computed using the following formulas:

$$q_f = K_z\, G_f\, q_{30} \qquad (3.3\text{-}2a)$$

$$q_p = K_z\, G_p\, q_{30} \qquad (3.3\text{-}2b)$$

where

K_z = the velocity pressure coefficient which depends upon the type of exposure and height z above ground, and

G_f, G_p = gust factors which depend upon the type of exposure and dynamic response characteristics of the structure, or parts and portions thereof.

Tables 5 and 6 of ANSI A58.1-1972 (Reference 1) provide the values of the effective velocity pressures for exposure C.

The design wind pressure P is calculated by

$$P = qC_p - q_m C_{pi} \qquad (3.3\text{-}3)$$

where q equals q_f or q_p whichever is appropriate, C_p is the external pressure coefficient, q_m is the effective velocity pressure for calculating internal pressures and C_{pi} is the internal pressure coefficient.

The external and internal pressure coefficients, C_p and C_{pi}, recommended in ANSI A58.1-1972, are summarized in Table 3.3-1.

For the containment wall and the dome, ASCE paper no. 3269, "Wind Forces On Structures" (Ref. 4), has been used to obtain the wind pressure coefficients as shown in Figure 3.3-2.

3.3.2 Tornado Loadings

3.3.2.1 Applicable Design Parameters

Clinton Power Station is located in Region I, as defined by Regulatory Guide 1.76.

The following are the parameters of the design-basis tornado (DBT):

a. a maximum wind speed of 230 mph

b. a maximum tangential velocity of 184 mph at a radius of 150 feet,

c. a maximum translational velocity of 46 mph,

d. an external pressure drop of 1.2 psi at a rate of 0.5 psi/s acting upon fully enclosed areas, and

e. a spectrum of tornado-generated missiles and their pertinent characteristics, as delineated in Subsection 3.5.1.4.

3.3.2.2 Determination of Forces on Structures

All tornado wind pressure and differential pressure effects are considered static in application, since the natural period of the building structures and their exposed elements is short compared to the time rate of the applied design pressure.

Venting has not been adopted as a design measure for Category I buildings. (Q&R 220.02)

The effects of the maximum credible tornado are translated into forces on the structures with the aid of a tornado model (Reference 3) that incorporates the design parameters of Subsection 3.3.2.1. The distribution of wind velocity with cyclonic radius for this model is represented by the following expressions:

$$\text{For } \frac{r}{R_C} \leq 1.0 \tag{3.3-4}$$

$$V(r) = V_C \frac{r}{R_C} + V_t$$

$$\text{For } \frac{r}{R_C} \geq 1.0 \tag{3.3-5}$$

$$V(r) = V_C \frac{R_C}{r} + V_t$$

where:

r = Radius from the center of vortex (ft),

V(r) = Tangential wind velocity at a radius r (mph),

R_C = Radius to the maximum wind velocity (ft),

V_C = Maximum tangential wind velocity (mph), and

V_t = Translational wind velocity (mph).

The distribution of the pressure drop as a function of the cyclonic radius for the model is represented by the following expressions:

$$\text{For } \frac{r}{R_C} \leq 1.0 \tag{3.3-6}$$

$$p(r) = P_C\left[1 - 0.5\left(\frac{r}{R_C}\right)^2\right]$$

$$\text{For } \frac{r}{R_C} \geq 1.0 \tag{3.3-7}$$

$$p(r) = \frac{P_C}{2}\left(\frac{R_C}{r}\right)^2$$

where:

p(r) = Pressure drop (psi) at radius r, and

P_c = Pressure drop at center of vortex.

All other terms previously defined.

The distribution of the differential pressure and tangential velocity plus translational velocity as a function of the distance from the center of the tornado, as represented by the above expressions, is shown graphically in Figure 3.3-3.

The tornado velocity is converted into an equivalent static pressure, using the equation 3.3-2. It is assumed there is no variation in velocity with height and that the gust factor is unity.

The pressure coefficients for windward pressure and for leeward, sidewall and roof suction given in Table 3.3-1 are used to determine tornado wind loading. The effective velocity pressure due to wind velocity alone is shown in Figure 3.3-4. Figure 3.3-5 shows the resultant static surface pressure when the pressure drop components and dynamic wind components are combined for rectangular flat-topped structures. For cylindrical and hemispherical structures the effective velocity pressure obtained from Figure 3.3-4 is used to obtain the pressure distribution shown in Figure 3.3-2 which is then combined with the pressure drop shown graphically in Figure 3.3-3 to obtain the resultant static surface pressure.

The tornado-generated missile loadings are considered as impactive dynamic loads. The method adopted for designing structures for this impactive load is described in Subsection 3.5.3.

The total tornado load is found by combining all three of the above-mentioned individual tornado loadings as follows:

i) $W_t = W_w$

ii) $W_t = W_p$

iii) $W_t = W_m$

iv) $W_t = W_w + W_p$

v) $W_t = W_w + W_m$

vi) $W_t = W_w + W_p + W_m$

where:

W_w = tornado wind pressure,

W_p = tornado differential pressure,

W_m = tornado missile load, and

W_t = total tornado load.

Depending on the particular structure under consideration, the most adverse effects of these combinations on the structure are used to derive the total design tornado load.

The total design tornado load is then combined with other loadings as per Tables 3.8-1 and 3.8-2.

When 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 intent of SRP No. 3.3.2 is to simplify the analysis and design of the structure for a tornado load based on the maximum wind velocity with the corresponding pressure drop without going into the detail of actual load distribution based on its location.

On the Clinton Project, the overall analysis of structures for the tornado load is based on the distribution of curves as shown in Figure 3.3-3. At any particular location the total tornado load is equal to the tornado load plus pressure drop. Therefore, this can be formulated as shown in the fourth load combination equation. The overall effect of the total tornado load on the structure is maximized by shifting its tornado center along the structure. Although the load combination equations for the total tornado loads used on the Clinton Project differ from those given in SRP Section 3.3.2.II.3(d), they are acceptable for determining the tornado loads for overall design of the structure.

The sixth load combination equation will be revised as follows based on the above explanation:

$$(\text{vi})\ W_t = W_w + W_p + W_m$$

For determining the tornado loads for the local design of structures on the Clinton Project, a maximum wind pressure plus its corresponding pressure drop (1.2 psi) as shown in Figure 3.3-3 were used. Although the load combination equations used for total tornado loads differ from those equations given in SRP Section 3.3.2.II.3(d), their numerical values are identical:

<u>SRP equation</u>

$$W_t = W_w + 0.5\ W_p + W_m$$

$$W_t = 232 + 0.5\,(3)(144) + W_m$$

$$W_t = 448 + W_m$$

<u>Clinton equation</u>

$$W_t = W_w + 1.0\ W_p + W_m$$

$$W_t = 232 + 1.0\,(1.5)(144) + W_m$$

$$W_t = 448 + W_m$$

W_m is the same in both equations. (Q&R 220.01)

3.3.2.3 Effects of Failure of Structures and Components Not Designed for Tornado Loads

The turbine building, radwaste building superstructure, circulating water screen house steel superstructure and service building are non-Seismic Category I structures. The turbine building is constructed on reinforced concrete up to elevation 800 feet 0 inch, and structural steel with metal roof decking and metal siding above elevation 800 feet 0 inch. The radwaste building superstructure is constructed of reinforced concrete. The service building is constructed of metal roof decking and precast concrete wall panels.

The turbine building siding and roof decking is designed to blow off in an approaching tornado, and the structure is designed to withstand tornado loads on the exposed structural frame and the remaining siding. The remaining siding is assumed to wrap around the center girts (see Figure 3.3-7). Thus, the integrity of the turbine building under the design-basis tornado is assured. The missiles generated in the blowoff of siding and roof decking of the turbine building are evaluated to be less damaging than the postulated tornado-generated missiles discussed in Subsection 3.5.1.4.

The steel superstructure of the circulating water screen house is not designed to withstand DBT. The Seismic Category I portions of this circulating water screen house are designed to withstand the effects of collapsing steel superstructure under DBT.

The vent stack on top of the diesel generator and HVAC building is designed to withstand DBT. The vent stack has been designed as a stiffened box girder to resist an equivalent static load due to the 360-mph wind speed of the design tornado with allowable stresses as delineated in Subsection 3.8.4.5. External missiles are considered by allowing them to penetrate the stack shell without impairing the overall integrity of the stack. (Q&R 220.04)

Turbine building siding is designed to blow in or out within a pressure range of 50 psf to 90 psf.

The turbine building siding is composed of double span blow-in/blow-out panels consisting of face and liner sheets interconnected by subgirts. There are no fasteners used at the lap joints between the adjoining face sheets to permit blow-in or blow-out (see Figure 3.3-6 sketch A).

Liner sheet is fastened to the girt at center support with screws penetrating the subgirt. At panel end supports, 3-inch wide, 12-gauge plate clamps down both liner ends. No screws penetrate the liner sheet (see Figure 3.3-6 sketch B).

Under wind pressure, because of large displacement, the liner ends slip out of the clamping plate and start to wrap around center support or start to bend away from center support for inward or outward pressure respectively (see Figure 3.3-7 sketches C and D).

After the siding panel collapses, it remains anchored to, and wrapped around the center support or remains anchored and bent outward for inward or outward pressure respectively (See Figure 3.3-7 sketches C and D).

Turbine building roof deck is designed to blow out within a pressure range of 55 psf to 70 psf for corners, and 40 psf to 60 psf for all other areas. This is assured through the use of pressure release fasteners. (Q&R 220.03)

The radwaste building superstructure has been designed for the SSE and tornado wind loads. Its failure is not deemed to be a credible design condition. The mass of the service building is small relative to that of the adjacent Category I structures. Therefore, its failure in the event of the SSE will not add sufficient lateral force to affect the integrity of the Category I Structures.

The CGCB has been designed to withstand the SSE. (Q&R 220.05)

3.3.3 References

1. American National Standards Building Code for Minimum Design Loads in Buildings and Other Structures, ANSI A58.1-1972, Section 6.

2. Design Basis Tornado for Nuclear Power Plants, USAEC Regulatory Guide 1.76.

3. J. D. Stevenson, "Engineering and Marketing Guide to Tornado, Missile Jet Thrust and Pipe Whip Effects on Equipment and Structures," Appendix B,:1-3, Report prepared for: Nuclear Structural Associates, Pittsburgh, Pennsylvania.

4. ASCE Paper No. 3269, " Wind Forces on Structures", Transaction of the American Society of Civil Engineers, Vol. 126, Part II, 1961.

TABLE 3.3-1
ANSI WIND PRESSURE COEFFICIENTS

ITEM	EXTERNAL COEFFICIENT(C_p)	INTERNAL COEFFICIENT(C_{pi})
Windward wall	0.8	+0.3
Leeward wall	-0.5	-0.3
Side wall	-0.7	-0.3
Roof	-0.7	-0.3
Corners	-2.0	-0.3
Cylinders and spheres	0.5	-0.3

3.4 WATER LEVEL (FLOOD) DESIGN

The Probable Maximum Flood (PMF) level elevation at the Clinton Power Station (CPS) site is 708.9 feet MSL. Refer to Subsection 2.4.3.5 for a complete description of this calculation.

For local intense Probable Maximum Precipitation (PMP) refer to Subsection 2.4.2.3. Compliance to Regulatory Guide 1.59 is discussed in Subsection 2.4.3.1.

For design purposes, the groundwater table at the CPS site is conservatively taken as elevation 730 feet. For actual groundwater elevations, refer to Subsection 2.4.13.

3.4.1 Flood Protection

3.4.1.1 Flood Protection Measures for Seismic Category I Structures

The effects of probable maximum flooding do not influence the station design because of the large difference in elevation between station grade (736 feet) and probable maximum flood elevation (708.9). However, the circulating water screen house is designed to withstand the effects of flooding.

The following protection measures are adopted for Seismic Category I systems and components located in the circulating water screen house and located below the probable maximum flood level. The flood protection arrangement of the Circulating Water Screen House is shown in Figure D3.6-134.

a. Water stops are provided in all construction joints up to the maximum flood level.

b. Water seal rings are provided for all penetrations in exterior walls below the maximum flood level.

c. Watertight doors designed to withstand the hydrostatic head of the maximum flood level are provided for all doorways located on both the entrance walls and the internal walls of the SSW pump rooms which are below the maximum flood level.

d. Hatches are provided on the roof of the essential service water pump structure (elevation 730 feet) for access during PMF.

The measures listed above are not required (except as noted below) at the CPS site, because grade at the station site is 27.1 feet above the PMF level. However, the measures are adopted for the portions of the structures at the station site located below the maximum groundwater level.

ECCS pump cubicles (RHR, LPCS, HPCS) located on the 707.5-foot elevation of the auxiliary and fuel (HPCS) building are protected from internal floods. All the walls, penetrations and doors in these cubicles are watertight to elevation 731 feet 5 inches (maximum flood level per Internal Flood Analysis). Watertight doors are shown on drawing M01-1105, sht 5. Also accounted for in the design is that flooding in one cubicle as a result of the rupture of a pump suction line from the suppression pool will not result in the flooding of other cubicles.

3.4.1.2 Permanent Dewatering System

The plant structures are designed to withstand the effects of groundwater conditions at the site. Therefore, a permanent dewatering system is not required and has not been installed.

3.4.2 Analysis Procedures

All substructures below elevation 730 feet 0 inch at the CPS site are designed to withstand full hydrostatic head of groundwater. The walls in the circulating water screen house that are exposed to floodwater are designed for hydrodynamic forces also.

The wind wave forces are determined in accordance with the "Shore Protection Manual," Volume II (U.S. Army Coastal Engineering Research Center, Department of the Army Corps of Engineering, 1973 and later Editions).

The hydrodynamic loads resulting from the seismic forces are determined in accordance with the procedures delineated in "Dynamic Pressures on Fluid Containers," Nuclear Reactor and Earthquakes, TID 7024, USAEC (August 1963).

The structural stability of Seismic Category I structures with flotation, overturning and sliding is investigated under the combined effects of the PMF and the wind wave forces. These criteria are the same as those given in Subsection 3.8.5.5.2. (Q&R 220.06)

3.5 MISSILE PROTECTION

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

All safety-related structures, systems and components in the auxiliary, radwaste, control, fuel, diesel generator and HVAC and circulating water screen house buildings are protected, to the extent practical, from the effects of postulated internal missiles. This is achieved by proper equipment layout where possible, otherwise suitable physical barriers are provided to isolate the missile or to shield the critical system or component.

Missile selection is done for pressurized as well as rotating type equipment. For pressurized equipment the following potential missiles are investigated:

a. valve bonnets (large and small),

b. valve stems,

c. thermowells, and

d. pressurized vessel head bolts.

The pressurized missiles are discussed in detail in Section 3.5.1.7.

For rotating equipment which has a potential for being subjected to an overspeed condition in excess of design limitations the potential missiles investigated are as follows:

a. pump blade,

b. pump impeller,

c. small flanges, and

d. coupling bolts.

3.5.1.2 Internally Generated Missiles (Inside Containment)

All safety-related structures, systems and components within the containment are protected to the extent practical from the effects of postulated internal missiles. This is achieved by proper equipment layout where possible, otherwise suitable physical barriers are provided to isolate the missile or to shield the critical system or component.

The following potential missiles from pressurized equipment were investigated:

a. valve bonnets (large and small),

b. valve stems,

c. thermowells, and

d. vessel head bolts.

These pressurized missiles are discussed in detail in Section 3.5.1.7.

In addition, the potential missiles investigated for rotating equipment are from the following:

a. pump blades,

b. pump impellers,

c. small flanges, and

d. coupling bolts.

In all cases investigated, it was shown that the kinetic energy of the potential missile was contained by the strain energy capacity of the equipment casing. Thus, it is demonstrated that the postulated missile is contained within the equipment casing.

The most substantial pieces of NSSS rotating equipment are the recirculation pump and motor. This potential missile source is covered in detail in Reference 9.

It is concluded in Reference 9 that destructive pump overspeed can result in certain types of potential missiles, but no damage is possible to any safety-related equipment because these missiles would not escape from the interior of either the pump or the motor.

With regard to evaluation of the probabilistic consequences of pump impeller missiles ejected from pipe breaks, it is concluded in Attachment 3 of Reference 9 that no damage is possible to primary containment, any major piping system, or an inboard main steam isolation valve. Absence of damage is because trajectories of postulated missiles do not interact with these systems.

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

The following potential missiles due to gravitational forces have been investigated:

a. Systems, components, and structures classified as seismic Category I are designed to withstand the applicable dynamic loads without failing. Therefore, they are not considered potential gravitational missile sources.

b. Non-seismic Category I items inside the containment are designed as follows:

    1. Structural Non-Category I classified items are designed not to fall; therefore, they cannot become a gravitational missile during reactor operation and following a LOCA.

    2. Lighting fixtures which are Non-Category I classified will be restrained in areas of safety-related items.

3. Non-Category I classified piping and piping supports have been designed to assure that the Class D piping and other non-safety piping will not fall or interact to impair the capability of the essential systems to perform their intended functions.

4. HVAC ductwork and hangers located in the Containment Building have been designed seismically with minor exceptions where failure of the duct system could not affect safety-related components.

5. Mechanical Non-Category I classified items such as unit heaters, area coolers, and HVAC instrumentation are designed seismically and, therefore, not considered gravitational missiles.

Subsections 3.5.1.1 and 3.5.1.2 have been supplemented by responses given to Grand Gulf Questions 211.13, 211.14, 211.15, and 211.16.

We have evaluated the effect of postulated missiles, blades, impellers, and shafts generated by typical rotating equipment such as pumps, fans, compressors, and turbines. It was concluded that in the unlikely event that internal missiles were generated, they would be contained by the equipment housing. One exception to the above protection analysis is certain fans supplied by Buffalo Forge Company, because calculations show that missiles could escape their housings. To protect essential equipment from internally generated missiles, fan housing reinforcement to prevent missile penetration, was provided for fans IVT06CA, IVT06CB, IVR04CA, IVR04CB, OVA05CA, OVA05CB, OVL02CA, OVL02CB, 1VF03CA, IVF03CB, OVA04CA, OVA04CB, OVW03CA, OVW03CB, IVR03CA, IVR03CB, IVT03CA, and IVT03CB.

Similarly we postulated missiles such as nuts, bolts, and valve stems from failure of pressurized components and conclude that the energy content of such missiles would be insufficient to cause damage or failure to safety-related equipment, components, or structures. (Q&R 410.1)

3.5.1.3 Turbine Missiles

The Nuclear Regulatory Commission (NRC) requires consideration of the effects of turbine missiles on the operation of nuclear power plants. A probability-based analysis is used to demonstrate protection against the effects of turbine missles at CPS.

A proprietary GE report to the NRC dated January 1984 and entitled, "Probability of Missile Generation in General Electric Nuclear Turbines," (NUREG 1048, Appendix U) described GE's methodology for evaluating the probability of wheel missle generation for nuclear turbines manufactured by GE. The methodology includes consideration of the probability of unit overspeed, wheel materials, in-service inspection capabilities and the potential for wheel containment by stationary turbine structures. The analysis methodology considers two fundamental failure modes that can lead to missile generation, brittle fracture failures and ductile tensile failures. The turbine originally supplied to CPS utilized conventional built-up rotors with shrunk-on wheels and axial keyways. The original built-up rotors have been replaced with rotors manufactured from monoblock forgings. The following is a discussion of the two fundamental failure modes and their applicability to the monoblock rotors for CPS:

Brittle Fracture Failure-

For the original built-up rotors with shrunk-on wheels operated in the speed ranges considered by GE, the probability of bursting, and thus of missile generation, is dominated by the brittle fracture mechanism. However, the replacement rotors for CPS are of monoblock construction and do not have shrunk on wheels. Therefore, the formerly dominant brittle fracture failure mechanism is not applicable to the new rotors.

Ductile Failure-

The probability of ductile failure for a rotor of any type is considered to be a function of speed, temperature and material tensile strength. With stress below ultimate strength, the probability of a ductile failure is negligible. The brittle and ductile failure modes are statistically independent.

The GE probabilistic analysis of turbine overspeed was also documented in the 1984 NRC report, and is applicable to units with LP monoblock rotors. The overspeed analysis considers the characteristics of the turbine control system, the unit configuration, and test requirements for the steam valves and other overspeed protection devices. This overspeed analysis showed that the probability of attaining a given overspeed decreases rapidly as the overspeed increases. As long as the control system is maintained in accordance with GE's recommendations (discussed below), the annual probability at CPS of attaining an overspeed of 120% or greater is 1.7×10^{-6}.

To keep the probability of a significant overspeed event very low, periodic maintenance and inspection of valves and other overspeed protection components are required. The intervals are established to maintain system reliability. The 1.7×10^{-6} per year probability assumes the longest permissible interval between valve inspections and would be lower with more frequent inspections.

The NRC has developed guidelines that limit the maximum annual probability for various hypothetical events. In the case of CPS, the limit for the annual probability for generation of a turbine missile is 1×10^{-4} (NUREG 1048, Appendix U, Table U-I). Since the CPS probability estimate of 1.7×10^{-6} is below the NRC threshold for probability of missile generation, protection against missile generation for the replacement CPS rotors can be shown by avoiding the potential for ductile failure at any operating speed below 120%.

GE has evaluated the tensile stresses in designing the rotating components of the monoblock turbine for CPS. All of the rotating components have sufficient margin to tensile strength at design component temperatures to support operating speeds well in excess of 120% of normal. For example, the overspeed capability of the un-bucketed HP and LP rotors is over 200%. The limiting components, per design, for the bucketed rotors are the LP L-0 buckets which have overspeed capability of 170%.

For the CPS rotors, the probability of attaining an overspeed of 120% is at or below 1.7×10^{-6} per year and there is a negligible probability of ductile failure at 120%. Therefore, the probability of turbine missile generation caused by ductile failure is well below the maximum NRC probability of $1 \text{ X } 10^{-4}$ per year and may be ignored.

3.5.1.4 Missiles Generated by Natural Phenomena

Tornadoes are the only natural phenomenon occurring in the vicinity of CPS that can generate missiles. Tornado impact velocities are obtained from the Tennessee Valley Authority Report TVA-TR74-1. The mathematical model in this report is developed on Hocker's observation of the Dallas tornado of April 2, 1957. The parameter values of this model are consistent with those of the design-basis tornado (DBT) specified in Regulatory Guide 1.76. The tornado missile trajectory is calculated by solving the equations of motion with the assumptions that (1) a potential missile does not tumble during the short period of time between its at-rest position and the point of its injection into the tornado wind field, and (2) the missile moves in a tumbling mode beyond its point of injection. The object that eventually becomes a tornado missile is assumed to be injected into the tornado wind field in the aerodynamic mode. The sample of missiles covers the wide spectrum of objects which might be windborne by a tornado and is listed in Table 3.5-3. The impact velocities of these missiles resulting from the design basis tornado (see Subsection 3.3.2) are shown in Table 3.5-3. Missiles A, and B, are considered at all elevations, and missile C is postulated at elevations up to 30 feet above grade level. These missiles are assumed to be capable of striking in all directions.

Table 3.5-5 lists (in general terms) the protection systems and/or component and the barrier for its protection from tornado missiles. Table 3.5-6 lists the structures/barriers, the concrete thickness strength, and the curing time on which the strength is based.

3.5.1.5 Missiles Generated by Events Near the Site

Based on a review of the nearby industrial, transportation, and military facilities (as described in Section 2.2), it is concluded that there are no potential missiles resulting from accidental explosions in the vicinity of the site.

3.5.1.6 Aircraft Hazard

The airports and airways in the vicinity of CPS are described in Subsection 2.2.2.5. The aircraft hazard does not constitute a design-basis event because of the following reasons:

a. There are no airports within 10 miles with projected operations greater than $500d^2$, nor are there any airports outside of 10 miles having the number of operations per year greater than $1000d^2$ (where d = distance in miles from the plant).

   Only two airports are known to accommodate commercial aircraft, the Decatur and Bloomington-Normal Airports.

   The Bloomington-Normal Airport is 22.5 miles north northwest of the plant and is served by American Eagle and American Flagship. Aircraft common to this airport are the Jetstream 31, Saab 340, DeHaviland 8, ATR 41 and 72, private jets, helicopters, and general aviation. Occasionally, charter flights will bring in Boeing 727 and 737, and McDonnell-Douglas DC-9. The number of operations (takeoffs and landings) during the 1994 calendar year was 89,457.

   The Decatur Airport is 24 miles south of the plant. It is normally served by American Eagle private and military helicopters, private jets and general aviation. The number of operations during the 1994 calendar year was 61,212.

The Rantoul National Aviation Center, formerly the Chanute Air Force Base (37 miles east northeast of the plant) is a municipal airport with an I5 (instrument classification) rating with daily flights of Citation 5s through general aviation. There are no regular commuter flights at this time.

b. Two of the private airstrips within five miles of the station (see Subsection 2.2.2.5) are:

Martin Airstrip located about 4.5 miles south of the station, and Thorp Airstrip located about 4.75 miles northwest of the station. Information obtained from the owners indicates that both airstrips are used for personal use only, and if needed for emergency. Martin has one turf runway 2,000 feet long oriented north-south and averages about 4 to 6 operations per week. Thorp has two grassy runways 1,500 feet long each, oriented north-south and east-west, and averages about 4 to 5 operations per week on each runway.

There are no commercial flights at these airports (Aircraft Owners and Pilots Association, 1975).

The Springfield office of the Division of Aeronautics states that the four airports within a 10-mile radius of CPS are small private airstrips that have had no recorded accidents in the last two decades.

The effective area of Category I Buildings is 0.00842 square miles and the probability of a fatal crash is taken as 1.2×10^{-8} per square mile per aircraft movement (SRP 3.5.1.6). Using the procedures in SRP 3.5.1.6, the probability of an aircraft crashing into the Category I Buildings is 0.315×10^{-7} per year for the Martin Airstrip and 0.525×10^{-7} per year for the Thorp Airstrips. The total impact probability by aircraft from the two airstrips is 0.42×10^{-7} per year, per unit, which meets the acceptance criteria of SRP 2.2.3.

c. The four low altitude federal airways described in Subsection 2.2.2.5 are:

Airway V313 with an average daily traffic of 20 flights;
Airway V233 with an average daily traffic of 20 flights;
Airway V434 with an average daily traffic of 15 flights; and
Airway V72 with an average daily traffic of 10 flights.

The actual width of each of these five airways is 8 nautical miles, or 9.21 statute miles.

Calculation of the probability of impact on the station by aircraft flying along these airways required evaluation of inflight crash rates (C), projected number of flights per year (N), effective area of the station (A), and width of airway (plus twice the distance from the airway edge to the station when the station is outside the airway) (W). As recommended by SRP 3.5.1.6 for light (less than 100 flights per day) commercial traffic, a value of $C = 3 \times 10^{-9}$ per aircraft mile has been used for each airway. The values used for N account for an increase in air traffic of about

89% during the 40 years life expectancy of the station. The 89% increase over the 40 year period is equivalent to the 21% increase in air carrier aircraft operations in FAA Aviation Forecasts for the 1980-1992 period. The effective area (A) used in the calculations include the actual plant area, a shadow area upon horizontal plane and a skid area of and around the Category I Buildings. The shadow area and the skid area and consequently the effective area depend on the orientation of the airways with respect to the buildings of the station. The distance from center of the station, the present yearly traffic (Np), the values of N, A, W, and the probability of impact (P_{FA}), on the Category I Buildings obtained for each airway are in Table 3.5-7. The sum of the four individual probabilities is 0.54×10^{-7} per year per unit, which meets the acceptance criteria of SRP 2.2.3.

d. There are three Guard aviation units within about fifty miles of the plant. There is an Army National Guard unit at the Decatur Airport (24 miles south) with an assortment of ten helicopters. There is an Air National Guard unit at the Springfield Airport (52 miles southwest) with F-16 fighters. And there is another Air National Guard unit at Peoria Airport (58 miles northwest) with C-130 aircraft.

3.5.1.7 Internally Generated Missiles From Pressurized Components

The station has been designed to assure that in the event of an internally generated pressurized missile:

1. There is no loss of containment function.
2. The reactor coolant pressure boundary is not breached as a result of postulated missile generation.
3. There is no loss of function to systems required to shut down the reactor and maintain it in a safe shutdown condition or to mitigate the consequences of such a postulated event:
    a. No equipment in one safety-related division is allowed to be damaged by a missile generated by equipment in another division.
    b. Missiles generated from non-safety related equipment shall not damage any safe shutdown equipment.
    c. Loss of offsite power is assumed concurrent with missile generation.

Those systems required to perform safety functions for this event are listed in Table 3.6-1.

A system is considered capable of generating a missile if, during normal plant conditions as defined in Subsection 3.6.1.1.1.d, it meets either of the following criteria for more than 2% of the time that the system is in operation:

1. The maximum operating temperature exceeds 200° F.
2. The maximum operating pressure exceeds 275 psig.

Those systems capable of missile generation are shown on Table 3.6-2. The sections of those systems capable of missile generation are further defined as the "dotted" lines in Figure 3.6-1.

The methods used to protect safety-related equipment from potential missiles generated internally due to pressurized components are as follows:

1. Provide design feature on pressurized equipment to prevent missile generation.

2. Locate high energy systems in separate missile-proof rooms.

3. Locate redundant system or components outside of the missile range and trajectory.

4. Orient the potential missile source to prevent unacceptable consequences of missile generation.

The following is the method used to select potential missiles generated by high energy lines as defined above:

1. Thermometers or other detectors installed on piping or in wells are evaluated as potential missiles if the failure of a single circumferential weld would cause their ejection.

2. Unrestrained sections of piping such as vents, drains, and test connections are evaluated as potential missiles if required due to postulated pipe breaks shown in Figures B3.6-1 through B3.6-34.

3. Valves of ANSI 900-pound standard rating and above and some valves of ANSI 600-pound standard rating, constructed in accordance with the ASME Code, Section III, are pressure seal bonnet-type valves. For these valves, the bonnets are prevented from becoming missiles by the retaining ring, which would have to fail, and by the yoke, which would capture the bonnet or reduce bonnet energy.

   Because of the highly conservative design of the retaining ring of these valves, bonnet ejection is highly improbable. Hence, bonnets are not considered credible missiles.

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

5. Single nuts, bolts, nut and bolt and nut and stud combinations are not credible missiles since all the valve body to bonnet connections are bolt through connections with a nut on one side. Also, all studs and bolts on primary system pressure boundary equipment are designed to ASME Section III standards and

are torqued to the stress levels allowed by ASME Section III. At these stress levels, the stored strain energy in the studs or bolts is too low to permit the stud or bolt to become a missile.

6. Valve stems are not considered as potential missiles if at least one feature in addition to the stem threads is included in their design to prevent ejection. Valves with backseats are prevented from becoming missiles by this feature. In addition, air and motor operated valve stems are effectively restrained by the valve operators. No credible valve stem missiles were identified on Clinton.

7. Dead end flanges are evaluated as potential missiles if required due to postulated pipe breaks shown in Figures B3.6-1 through B3.6-34.

Our analysis of thermowells shows that the energy associated with either a piston type or jet propelled type missile is low and will not cause damage to any essential components.

Piston-Type Missile The velocity of a piston-type missile (e.g., valve stem, etc.) is calculated by assuming that 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 produce of missile mass times the square of the missile velocity. Assuming the force constant (and equivalent to PA_O) and equating the kinetic energy to the work done results in a missile velocity given by the expression (Reference 10).

$$V = \left[\frac{2PA_O I}{w/g} \right]^{1/2} \qquad (3.5\text{-}6)$$

where

P	=	Pressure acting on area A_O (lb/ft^2)
A_O	=	Area of missile under pressure (ft^2)
I	=	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)

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 employ in the jet property solution as given by Moody (Reference 11) for saturated steam blowdowns.

The work of Reference 11 was directed toward the prediction of blowdown thrust and jet forces on stationary targets; however, by making a few simplifying assumptions and apply in 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:

$$\frac{y}{(W/A)} = V_{\infty}\left[\ell n\left(\frac{1}{1-V/u_{\infty}}\right)-\frac{V}{u_{\infty}}\right] \qquad (3.5\text{-}7)$$

where

W/A

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/lbm)
V	=	Velocity of missile (ft/sec)

Neglecting friction we used equation 3.5-7 by expanding it in the power series neglecting higher order terms to solve for velocity of the missile. Missile velocities are calculated at three regions: (a) at the break, (b) at the asymptote or transition area, (c) at the fully developed flow region. These are calculated based on specific stagnation properties (temperatures and pressures).

Those areas of the plant where high energy lines were located within the same room as safety-related components required for safe shutdown are identified in Table 3.5-8. These are the only areas of the plant where internally generated missiles could potentially impact essential components.

In the control building, the fuel building and auxiliary building identified in Table 3.5-8, there are no potential missile sources from high energy lines which could impact essential components. All high energy lines in the remaining areas of the plant, the steam tunnel, the containment and the drywell were reviewed for potential missile sources. These potential missile sources and all the targets that they could hit are listed in Table 3.5-9. As shown in Table 3.5-9, no potential missile could hit and disable components which are essential for plant safe shutdown.

Therefore, the design of the Clinton Power Station incorporates sufficient separation between divisional equipment to assure that the containment can be isolated, the reactor coolant boundary is not breached, and the reactor can be safely shut down even with a loss of offsite power.

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

3.5.2.1 General

Missile selection and description for those external missiles which, if generated, could damage plant structures, systems or components important to safety, are identified in Subsections 3.5.1.4 and 3.5.1.5.

3.5.2.2 Structures Providing Protection Against Externally Generated Missiles

Seismic Category I structures are designed to withstand postulated external missiles, thereby protecting the systems and components located within. Openings in the Unit 1-Unit 2 (Unit 2 has been cancelled) interface walls of Category I structures are closed with tornado missile resistant concrete barriers. These concrete barriers are considered part of the Category I structure, and the missile proof walls are shown in Figure 3.5-3. Protective characteristics of Seismic Category I Structures are summarized in Table 3.5-6.

3.5.2.3 Barriers (Other Than Structures) Providing Protection Against Externally Generated Missiles

Those structures, systems and components to be protected from externally generated missiles, and the missile barrier associated with each, are identified in Table 3.5-5. The missile barriers indicated are designed for tornado-generated missiles using the procedures given in Subsection 3.5.3. Structures which protect plant systems and components from missiles generated outside the plant are identified in Subsection 3.5.2.2.

Protection for those safety related systems and components not located within Seismic Category I structures (i.e. outdoors) is also identified in Table 3.5-5.

The location of missile barriers is shown in Figures 3.5-3 through 3.5-5.

3.5.2.4 Systems/Components Not Requiring Unique Tornado Missile Protection

A limited amount of safety related systems and components located near penetrations in Seismic Category I structures or located outside of such structures are evaluated as not requiring unique tornado missile protection barriers. Two approaches were used in the evaluation (reference 15):

1. Certain safety related systems and components are screened out using the criteria of Regulatory Guide 1.117, Tornado Design Classification, including its Appendix, which together, detail the items that should be protected from the effects of tornadoes. The criteria in the Regulatory Guide are summarized as important systems and components required to ensure the integrity of the reactor coolant pressure boundary, ensure the capability to shut down the reactor and maintain it in a safe shutdown condition, and those whose failure could lead to radioactive releases resulting in calculated offsite exposures greater than 25% of the guideline exposures of 10 CFR Part 100 using appropriately conservative

analytical methods and assumptions. The safety related systems and components not required to support these Regulatory Guide 1.117 guidelines are evaluated as not requiring unique tornado missile protection.

2. "Important" systems and components (as discussed in Regulatory Guide 1.117) are generally protected. The limited amount of unprotected portions of important systems and components are analyzed using a probabilistic missile strike analysis as permitted in Standard Review Plan 3.5.1.4, "Missiles Generated By Natural Phenomena". This analysis is conducted to determine the total (cumulative) probability per year of missiles striking important structures, systems, and components due to postulated tornadoes. This information is then utilized to determine the specific design provisions that must be provided to maintain the estimate of strike probability below an allowable level.

The allowable level established for the protection of such systems and components at CPS is consistent with the acceptance criteria in Standard Review Plan 2.2.3, "Evaluation of Potential Accidents", i.e., that a probability of occurrence of initiating events (those that could lead to potential consequences in excess of the 10 CFR Part 100 guidelines) of approximately $1x10^{-6}$ per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower. The CPS-specific acceptance criteria is that the total probability of tornado missiles simply striking an important system or component must be shown by analysis to be less than $1x10^{-6}$ per year.

This CPS-specific criteria contains the following inherent conservatisms:

- It is assumed that an important system or component simply being struck by a tornado missile would result in damage sufficient to preclude it from performing its intended safety function, although this is not realistic for all cases.
- The analysis calculates the probability of tornado missiles striking penetration openings. The openings themselves are not targets. The true targets are the safety-related components located inside the buildings. Some of the missile types listed in Table 3.5-3 cannot enter the openings and damage the components.
- The missile population is conservatively estimated.
- All postulated missiles are conservatively estimated to have minimal restraints.

The analysis uses an NRC approved methodology (Reference 13) developed by the Electric Power Research Institute (EPRI)(Reference 14). The methodology is implemented using the computer program TORMIS, which is described in Section 3.5.2.5.

Should CPS evaluations using the TORMIS methodology provide results indicating that the probability of tornado damage exceeds the acceptance criteria of $1x10^{-6}$ per year, then unique barriers are utilized to reduce the total probability to below the acceptance criteria. Temporary removal of a protective feature is permitted under administrative controls, if removal is determined to be necessary.

3.5.2.5 <u>TORMIS Description</u>

TORMIS implements a methodology developed by the Electric Power Research Institute. TORMIS determines the probability of striking walls and roofs of buildings on which penetrations or exposed portions of systems/components are located. The probability is calculated by simulating a large number of tornado strike events at the site for each tornado wind speed intensity scale. After the probability of striking the walls or roof is calculated, the exposed surface area of the particular components are factored in to compute the probability of striking a particular item.

The TORMIS analysis for the CPS site (Reference 15) is in accordance with the TORMIS program, as described in Reference 14, using site-specific parameters described below:

1. The probability of a tornado strike at CPS is based upon the broad region values associated with the Fujita F-scale.

2. The Fujita scale (F-scale) wind speeds are used in lieu of the TORMIS wind speeds (F-scale) for the F_0 through F_5 intensities. In addition, a wind speed range from 320 to 360 mph is used for the F_6 intensity to correspond to the tornado wind speed described in Section 3.3.2.1.

3. A more conservative near-ground profile was used than the base case in TORMIS, resulting in a higher tornado ground wind speed. The profile has a ground wind speed equal to 82% of the wind speed at 33 feet (i.e., $V_0/V_{33}=0.82$).

4. The number of missiles used in the CPS TORMIS analysis is a conservative value for CPS-specific sources, such as laydown, parking, and warehouse areas. These are postulated by general walkdown information at CPS.

3.5.3 <u>Barrier Design Procedure</u>

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

a. Local effect in the impacted area which includes estimation of the depth of penetration and, in the case of concrete barriers, the potential for secondary missiles by scabbing.

b. Overall response of the barrier, which includes the calculation of deflection due to missile impact.

Generally, all missiles (internal or external) are considered as impacting instantaneously with a very short rise time relative to the natural period of the impacting structure. Two types of barriers are designed to resist missile impact, as follows:

a. Reinforced concrete barriers: The depth of penetration into a concrete barrier is calculated using either the modified Petry equation (Reference 5) or the modified NDRC formula (Reference 12). The material property constant used in the penetration formula is 4.76×10^{-3} ft^3/lb. Concrete barriers are designed such that the missile penetrates no more than two-thirds of the thickness of the barrier, preventing scabbing (Reference 6). The overall deformation of the panel is investigated using methods presented in Reference 6. Reference 6 presents an equation of motion which makes it possible to calculate an impact force time-history consistent with the calculated penetration depth. To establish the capacity of the barrier to absorb energy, the deflection due to static loads is first calculated. The deflection due to missile impact is then determined by integrating the equation of motion or by using a simplified expression adopted from the equation of motion. This is compared with the maximum allowable deflection (of allowable ductility ratio) per ACI 349.

Elements encased in concrete with 5 inches of cover and buried in 4 feet of soil are investigated using the soil penetration equations given in the reference. This investigation showed that most of the missiles listed in Table 3.5-4 did not penetrate the four feet of soil cover.

For those missiles with penetration depths of more than four feet, a striking velocity at the concrete surface is estimated based on energy balance, and a penetration depth into the reinforced concrete is calculated using Equation 3.5-5. The penetration depths thus obtained are found to be much less than the minimum avaiable depth of reinforced concrete cover.

For elements with less than the required amount of soil or concrete cover, the probability analysis was performed. The analysis showed that the probability of a tornado generated missile striking or damaging these elements is less than 1×10^{-7}. Therefore, the tornado missile is not considered a design basis for these elements.

<u>REFERENCE</u>:

Young, C. W., "Depth Prediction for Earth-Penetrating Projectiles," Journal of the Soil Mechanics and Foundations Division, ASCE, Vol. 95, No. SM3, Proc. Paper 6588, May 1969, pp. 803-817. (Q&R 220.11)

b. Steel plate barriers (Note) - The thickness of steel plate required to resist the impacting missile is calculated using the Stanford formula (Reference 7). The overall structure response, including structural stability and deformation, is investigated using concepts and methods presented in Reference 8.

The only steel barriers provided are those employed for tornado missile protection of pipe sleeves in exterior wall and roof. The design was based on the modified Stanford Research Institute (SRI) formula, as noted in subsection 3.5.3, with its inherent ductility requirements. The pipe sleeve caps are sufficiently similar to the test specimens used in developing the SRI formula; therefore, overall behavior is included in the results. (Q&R 220.10)

Note: Steel plate barriers in this section also include those of steel grating.

It is to be noted that the location of the reactor shield wall inside the containment structure is such that no potential missile will strike the reactor pressure vessel.

In Reference 6, the maximum displacement, Y_m, of a structural element under the impact of a rigid missile of mass m is given by:

$$Y_m = \frac{m}{M}\left(D'^2 + \left(\frac{V_o}{\omega}\right)^2\right)^{1/2} \qquad (1)$$

where:

M	=	equivalent mass of structural element
D'	=	penetration of missile into structure
V_o	=	missile velocity prior to impact
ω	=	ciruclar frequency of structural element

The method specified in SRP Section 3.5.3 for treating this problem is given in the reference cited at the end of this response. This reference specifies the equivalent static design load, F, as:

$$F = \frac{F_i}{K} \qquad (2)$$

$$F_i = m\left(\frac{V_o 2}{D'}\right) \qquad (3)$$

$$t_1 = \frac{2D'}{V_o} \qquad (4)$$

$$K = \frac{\sqrt{2\mu - 1T}}{\pi t_1} + \frac{1 - \dfrac{0.5}{\mu}}{1 + \dfrac{0.7T}{t_1}} \qquad (5)$$

where:

F_i	=	peak value of impact force developing between missile and target
t_1	=	duration of impact
T	=	$\dfrac{2\pi}{\omega}$
μ	=	ductility ratio = $\dfrac{Y_m}{\delta y}$

δy = static yield deflection, when element is idealized by an SDF dynamic system

To compare the two methods, a square reinforced conrete panel 26 feet long and 1.5 feet thick, with reinforcement details and material properties as summarized in the attached Figure 3.5-6 was considered. The results obtained by the two methods for the response of this panel to the impact of a rigid mass at the center are summarized below. The missile corresponds to the utility pole described in Tables 3.5-3 and 3.5-4 of the USAR, with

V_o = 241 ft/sec

$$m = \frac{1{,}500 \text{ lb sec}^2}{32.2 \text{ ft}}$$

Head-on concrete area = 143.10 in^2

For this missile, the modified Petry formula with K_p = 0.0032 ft^3/lb yields a missile penetration value of D' = 5.976 in.

The panel is approximately as an SDF system for response evaluation, with a mass of

$$M = 0.32\left(\frac{\text{panel weight}}{g}\right)$$

where g = acceleration gravity.

The period of the system is determined on the basis of a cracked section moment of inertia to be 0.0961 second. The yield force Q_y and the corresponding panel deflection δy are 358.5 kips and 0.680 inches, respectively.

The maximum displacement for Method 1 (the method given in the USAR) is obtained directly from Equation 1:

$$[Y_{max}]_{method\ 1} = 1.37 \text{ inches}$$

To obtain the maximum displacement for Method 2 (the method given in SRP Section 3.5.3), a trial and error procedure is used for Equations 2 through 5 to obtain the ductility parameter. This value is

$$[\mu]_{method\ 2} = 2.68$$

Therefore,

$$[Y_{max}]_{method} = 2.68 \times 0.0680 = 1.82 \text{ inches}$$

The two results are within 25% of each other. To provide an indication of the difference between the results from the two methods when other ductility ratios are used, the velocity of the missile is arbitrarily increased from 241 ft/sec to 364 ft/sec. This yields a missile penetration of 12 inches, which is 2/3 of the panel thickness (the maximum penetration permitted per the USAR). Under this condition,

$$[Y_{max}]_{method\ 1} = 2.08\ \text{inches}$$

$$[\mu]_{method\ 2} = 5.48$$

$$[Y_{max}]_{method\ 2} = 5.48 \times 0.680 = 3.73\ \text{inches}$$

The maximum displacements agree to within 44%.

It should be noted that in Equation 3 above, the work of the missile on the target during the penetration process has not been evaluated consistently with the linear variation of contact force and velocity with time, which is the starting point of derivation in the reference. When the equation is done consistently with these assumptions, Equation 3 changes to Equation 3*:

$$F_i = \frac{3}{4} m \frac{V_o 2}{D'} \qquad (3^*)$$

The values obtained for Y_{max} for the two cases described above then become

$$[Y^*_{max}]_{method 2} = 1.12 \text{ for } V_o = 241\ \text{ft/sec}$$

$$[Y^*_{max}]_{method 2} = 2.24 \text{ for } V_o = 364\ \text{ft/sec}$$

The ratios of $Y_{max_{method 1}} / Y_{max_{method 2}}$ are given below:

Missile Velocity ft/sec	Penetration Depth inches	Max. Deflection Ratio of the Two Methods
241	5.9	1.22
364	12	.927

Since the underestimation of method 1 is less than 8%, the use of this method is considered acceptable.

REFERENCE

(From SRP 3.5.3): R. A. Williamson and R. R. Alvy, "Impact Effect of Fragments Striking Structrual Elements," Homes and Narver, Inc., Revised November 1973. (Q&R 220.07)

3.5.4 References

1. NUREG 1048, Supplement 6, Appendix U, "Probability of Missile Generation in General Electric Nuclear Turbines," (Hope Creek SSER 6), 7/86.

2. Not used.

3. S. H. Bush, "Probability of Damage to Nuclear Components Due to Turbine Failure," Nuclear Safety, 14(3), pp. 187-201, May-June 1973.

4. Not used.

5. A. Amirikian, "Design of Yards Protective Structures," Bureau of Yards and Docks, Publication No. NAVDOCKS p-51, Department of the Navy, Washington, D.C., August 1950.

6. J. M. Doyle, M. J. Klein, and H. Shah, "Design of Missile Restraint Concrete Panels," 2nd International Conference on Structural Mechanics in Reactor Technology, Berlin, Germany, September 1973.

7. W. B. Cottell and A. W. Savolainen, "U.S. Reactor Containment Technology," ORNL-NSIC-5, Vol. 1, Chapter 6, Oak Ridge National Laboratory, August 1965.

8. R. A. Williamson and R. R. Alvy, "Impact Effect of Fragments Striking Structural Elements," Holmes and Naver, Inc., Revised November 1973.

9. GE Letter Report, "Analysis of the Recirculation Pump Under Accident Conditions," Revision 2, March 30, 1979.

10. R. C. Gualtney, "Missile Generation and Protection in Light Water Cooled Power Reactor Plants," USAEC Report ORNL-NSIC-22, September 1968.

11. F. J. Moody, "Prediction of Blowdown Thrust and Jet Forces," ASME Publication G9-HT-31, August 1969.

12. George E. Sliter, "Assessment of Empirical Concrete Impact Formulas," ASCE Journal, Volume 106, No. ST5, May 1980.

13. Letter, Rubenstein (NRC) to Miraglia (NRC) entitled "Safety Evaluation Report - Electric Power Research Institute (EPRI) Topical Reports Concerning Tornado Missile Probabilistic Risk Assessment (PRA) Methodology", dated October 26, 1983.

14. Twisdale, L.A. and Dunn, W.L., EPRI NP-2005, Tornado Missile Simulation and Design Methodology, Volumes I and II, Final Report dated August 1981.

15. Sargent & Lundy Report SAD-465, Revision 2, September 16, 1998, Tornado Missile Hazard Assessment for Clinton Power Station.

Table 3.5-1 Has Been Deleted

Table 3.5-2 Has Been Deleted

TABLE 3.5-3

TORNADO-GENERATED MISSILES AND THEIR PROPERTIES

MISSILE		WEIGHT (lb)	DIMENSIONS	CdA/m (ft^2/lb)	HORIZONTAL IMPACT VELOCITY (Note 1) (ft/sec)
A.	Solid Steel Sphere	0.147	1 in. diameter	0.0166	26
B.	6-inch Schedule 40 steel pipe	287	6.625-in. OD x 15 feet long	0.0212	135
C.	Automobile (Note 2)	4,000	16.4 ft x 6.6 ft x 4.3 ft	0.0343	135

Note 1. Vertical impact velocities are taken equal to 67% of the horizontal impact velocities.

Note 2. The automobile missile is considered to impact at all altitudes less than 30 feet above all grade levels within 0.5 mile of the plant structures.

Table 3.5-4 Has Been Deleted

TABLE 3.5-5 (Cont'd)

TABLE 3.5-5
PROTECTED COMPONENTS AND ASSOCIATED MISSILE BARRIERS
FOR EXTERNALLY GENERATED MISSILES

	COMPONENT	BARRIER
A.	Protected Components Within the Plant	
1.	Reactor coolant pressure boundary and other safety-related equipment inside containment	Containment structure, dry- well, internal structures, and beams
2.	Emergency core cooling, containment spray, cooling water, ventilation, electrical, instrumentation, control, and other safety-related equipment in auxiliary building	Containment building, auxiliary building, and internal structures
3.	Control room and protected electrical, instrumentation, control, and ventilation equipment in control building	Control building
4.	Spent fuel pool	Fuel pool walls, fuel building
5.	Emergency diesel generators and diesel fuel oil system (1)	Diesel-generator building
6.	Shutdown service water pumps and associated piping	SSW portions of the Circulating Water Screen House protects the SSW Pumps and associated piping. Some SSW piping is located beneath steel plates in construction openings which haven't been demonstrated to be complete missile barriers. The openings are analyzed in the TORMIS analysis, described in 3.5.2.4 and 3.5.2.5.
7.	Portion of the reactor coolant pressure boundary in the auxiliary building	Auxiliary building and auxiliary building steam tunnel
B.	Protected Components Outdoors	
1.	Electrical manholes (Category I)	Protected by a reinforced 1-foot-thick concrete cover with steel plate manhole covers (1-inch-thick galvanized plate)
2.	Electrical duct banks (Category I)	Protected by a minimum of 5 inches of reinforced concrete, buried a minimum of 4 feet below finish grade with exceptions described in Section 3.5.3, item a. Reinforced concrete barriers.

TABLE 3.5-5 (Cont'd)

	COMPONENT	BARRIER
3.	Control building	
	a. Ventilation air intakes	Control Building ventilation air intakes are protected by a minimum 2-foot-thick reinforced concrete missile barrier (see Figure 3.5-3) except for the VA and VR system air intakes which are protected by two layers of thick heavy duty grating. Other openings in the Control Building walls, which are not protected and are analyzed in the TORMIS analysis described in 3.5.2.4 and 3.5.2.5, are for unused Unit 2 air intakes.
	b. Ventilation air exhausts	Protected by a minimum 2-foot-thick reinforced concrete missile barrier (see Figure 3.5-3)
	c. External access doors	Doors are designed to withstand tornado missiles or they are protected by a minimum 2-foot-thick reinforced concrete missile barrier, except for steel roll-up door which is protected by two layers of thick heavy duty grating.
4.	Auxiliary building	
	a. Access doors	Access doors to the Auxiliary Building are internal to the Turbine Building, Control Building, and Fuel Building and are protected by these buildings.
5.	Diesel-generator building	
	a. Ventilation air intakes	Protected by a minimum 2-foot-thick reinforced concrete missile barrier (see Figure3.5-3)
	b. Ventilation air exhausts (for the diesel generator combustion air exhaust exception see Section 9.5.8.1.1.b.)	Protected by a minimum 2-foot-thick reinforced concrete missile barrier (see Figure 3.5-3)
	c. Standby gas treatment system exhaust	Barrier not required for those portions external to diesel-generator building

	COMPONENT	BARRIER
	d. Station common exhaust vent	Barrier not required because missile trajectory through this opening cannot endanger safety-related structures,systems or components
	e. Access doors	Doors are designed to withstand tornado missiles or they are protected by a minimum 2-foot-thick reinforced concrete missile barrier
	f. Diesel fuel oil storage tank fill lines	Protected by diesel-generator building except as noted in Subsection 9.5.4.3
6.	Fuel building	
	a. Access doors	Doors are designed to withstand tornado missiles or they are protected by a minimum 2-foot-thick reinforced concrete missile barrier

TABLE 3.5-6
CONCRETE BARRIER PARAMETERS

Structures	Minimum Concrete Thickness	Design Strength at 91 Days (ksi)
Auxiliary Building Walls	2'-0"	3.5
Auxiliary Building Roof	1'-6"	3.5
Fuel Building Walls	2'-0"	3.5
Fuel Building Roof	2'-0"	3.5
Control Building Walls	2'-0"	3.5
Control Building Roof	2'-0"	3.5
Diesel Generator Building Walls	2'-0"	3.5
Diesel Generator Building Roof	2'-0"	3.5
Containment Wall	3'-0"	4.0
Containment Dome	2'-6"	4.0
Circulating Water Screen House Walls	2'-0"	3.5
Circulating Water Screen House Roof	1'-6"	3.5

Note:

Penetrations in exterior walls and roofs of Safety Related buildings are analyzed using the TORMIS analysis described in USAR Sections 3.5.2.4 and 3.5.2.5.

TABLE 3.5-7
PROBABILITY OF AIRCRAFT IMPACT FROM FEDERAL AIRWAYS

AIRWAY	DISTANCE FROM THE STATION IN MI., d	PRESENT TRAFFIC PER YEAR, N_p	PROJECTED TRAFFIC PER YEAR $N = n_p x(1.21)\frac{40}{12}$	EFFECTIVE AREA IN SQ, MI., A (2 Units)	WIDTH IN MI., w	PROBABILITY OF IMPACT PER YEAR (2 Units) P_{FA}=CxNxA/w
V313	1.5	7,300	13,780	0.00842	9.21	0.378×10^{-7}
V233	2.0	7,300	13,780	0.00842	9.21	0.378×10^{-7}
V434	6.0	5,475	10,335	0.00690	12.0	$0.178 \times 10\text{-}^{7}$
V72	4.75	3,650	6,890	0.00690	9.5	0.150×10^{-7}
					TOTAL	1.084×10^{-7}

Probability of Impact Per Year Per Unit = 0.542×10^{-7}

NOTE: UNIT 2 HAS BEEN CANCELLED.

TABLE 3.5-8
AREAS WHERE ESSENTIAL COMPONENTS AND HIGH ENERGY LINES OCCUR TOGETHER

Drywell

Containment

Steam tunnel

Auxiliary building

Fuel building

Control building

TABLE 3.5-9
POTENTIAL MISSILE SOURCES WHICH
COULD IMPACT SAFETY RELATED COMPONENTS

POTENTIAL THERMOWELL MISSILE	LOCATION	REMARKS
1B33-N023A 1B33-N023B 1B33-N028A 1B33-N028B	Drywell	The reactor recirculation pump which is hit is not required for shutdown nor would it be pierced by the missile
1B33-N021 1B33-N022	Drywell	Hanger 1RT28007S – Not essential for shutdown Conduit C74523 – Will withstand missile Hanger 1RT01020S – Will withstand missile Weir wall – Will withstand missile
1G33-N004	Containment	FW Guard Pipe – Will withstand missile
1G33-N015	Containment	Valve 1G33-F031 and associated Flex Conduit - Not essential for shutdown. Valve itself will withstand missile
1G33-N007	Containment	Pipe 1RT11A4 - Will withstand missile
1G33-N019	Containment	Pipe 1RT06A6 - Will withstand missile Pipe 1RT05D4 – Will withstand missile Steam Tunnel Wall – Will withstand missile
1G33-N020	Containment	Pipe 1RT05D4 - Will withstand missile Steam Tunnel Wall – Will withstand missile
1G33-N006A	Containment	Hanger 1RE16033G – Not essential for shutdown Steam Tunnel Wall – Will withstand missile

TABLE 3.5-9 (Cont'd)

POTENTIAL MISSILE SOURCES WHICH
COULD IMPACT SAFETY RELATED COMPONENTS

POTENTIAL THERMOWELL MISSILE	LOCATION	REMARKS
1G33-N006B	Containment	Valve 1G33-F022E – Will withstand missile Hanger 1RE20049X – Will withstand missile Hanger 1RE20060S – Not essential for shutdown Steam Tunnel Wall – Will withstand missile
1B21-N040	Auxiliary Steam Tunnel	Hanger 1RI08020R – Will withstand missile
1B21-N059	Auxiliary Steam Tunnel	Pipe 1FW25AB 3/4 – Not essential for shutdown Valve 1B21-F466B – Not essential for shutdown Pipe 1RI24A2 – Will withstand missile
1B21-N060	Auxiliary Steam Tunnel	Pipe 1MS35A3 – Not essential for shutdown

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

Piping failures in normally operating high- and moderate-energy fluid systems are postulated. The direct results of a piping failure are pipe whip, fluid impingement, environment pressurization, temperature and humidity effects, and flooding. Coincident with the piping failure, the functional failure of any single active component, a seismic event the level of the safe shutdown earthquake and a loss of offsite power are assumed to occur.

Given the above, the safety function of essential systems and components will not be impaired beyond that required to bring the plant to a safe shutdown.

This section describes the design bases and protective measures which ensure that the containment, essential systems, components and equipment, and other essential structures are adequately protected from the effects associated with the postulated rupture of high-energy piping and cracks of high- and moderate-energy piping both inside and outside the containment.

3.6.1 Postulated Piping Failures in Fluid Systems

The following is a summary of applicable definitions, criteria employed, potential sources and locations of piping failures, identification of systems and components essential to safe plant shutdown, limits of acceptable loss of function or damage and effect on safe shutdown, habitability of critical areas following postulated piping ruptures, and the impact of the plant design on inservice surveillance and inspection.

The steam tunnel subcompartment pressure analysis inside containment is given in Subsection 6.2.1.2. The results of the steam tunnel subcompartment pressure analysis outside containment are presented in Subsection 3.6.1.2.2 and Table 3.6-3. The details of the steam tunnel pressure analysis outside containment are presented below.

The steam tunnel outside containment is subject to pressure differentials when a high energy line with the tunnel is postulated to rupture. A simultaneous break of one main steamline and one feedwater line was considered in this analysis to determine the accident conditions within the steam tunnel.

The steam tunnel is a passageway starting at the primary containment boundary, which is dead-ended, and the ending in the turbine building. It is made of two sections; one section is horizontal run which starts at the containment boundary and ends in the turbine building, and the other section is a vertical section located in the turbine building (Drawings M01-1107, M01-1111 Sheet 1, and Figure 6.2-132 Sheets 3, 4, and 6.)

The only exit from the steam tunnel is into the turbine building at the grade floor (elevation 737 feet 0 inches).

All of the lines present in the steam tunnel are considered to be high energy lines. There are four main steamlines and two feedwater lines. The simultaneous break of one main steamline and one feedwater line was assumed to be an instantaneous guillotine rupture within the steam tunnel. The pipe breaks inside the tunnel were postulated to occur approximately 10 feet downstream of the outer main steam isolation valves since this results in the most severe pressure transient in the tunnel. General Electric data on mass and energy release for one main steamline downstream of the outer main steam isolation valves was used to determine the break flow for the main steamline. The conditions assumed in the feedwater line were 1,077 psig, 430°F and an enthalpy of 408.5 Btu/lbm. The feedwater line break mass and energy releases were also evaluated at a reduced feedwater temperature (RFWT) of 380°F and an enthalpy of 354.8 Btu/lbm.

It was also conservatively assumed that the feedwater flowed out of the break with sonic velocity for the entire period of the transient. The feedwater break flow was calculated using the Moody critical flow model. The feedwater flow was assumed to be limited from the reactor side by an instantaneous closure of the check valve in the line, but unimpeded from the heater side. For conservatism, it was assumed that only one check valve located in the drywell operates under accident conditions so that approximately 87 cubic feet of feedwater volume (in 62 feet of pipe of 1.407 square feet flow area) was available from the reactor side for flow out of the feedwater break. The mass and energy release rates are shown in Table 3.6-7.

The nodalization of the steam tunnel analysis required the nodalization of the steam tunnel, the basement, grade floor, mezzanine floor, and the turbine floor of the turbine building. In addition, another node was required to model the atmosphere outside the turbine building. The nodalization is shown in Figure 3.6-15. It was calculated that the turbine room siding on the turbine floor of the turbine building would yield if the pressure on the turbine floor approached 15.4 psia. This would open a flow path equal to 2,000 square feet between the turbine floor and the atmosphere. The volume and vent path characteristics were determined in a similar manner as described in Subsection 6.2.1.2.3.2.1 and all of the nodal and vent path descriptions are shown in Tables 3.6-8 and 3.6-9, respectively. The initial conditions within the steam tunnel were assumed to be at 14.7 psia and a relative humidity of 100%.

The initial temperatures within the steam tunnel nodes were assumed to be 150°F, and the initial temperatures within the turbine building and atmospheric nodes were assumed to be 104°F.

The simultaneous rupture of one main steamline and one feedwater line within the steam tunnel resulted in a maximum pressure of 13.8 psig, 0.25 seconds after the initiation of the accident. The pressure history is shown in Figure 3.6-16.

The first Feedwater Isolation Valve outside the containment within the steam tunnel is designed as a Class 1 valve for severe duty application with environmental effects more severe that the accident conditions derived from the simultaneous break of one main steamline and one feedwater line.

Information on the main steamline isolation valves is given in Subsection 6.2.4.2, System Design. (Q&R 410.2)

3.6.1.1 Design Bases

3.6.1.1.1 Definitions

a. Essential Systems and Components

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

b. High-Energy Fluid Systems

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

1. Maximum operating temperature exceeds 200° F.

2. Maximum operating pressure exceeds 275 psig.

c. Moderate-Energy Fluid Systems

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

1. Maximum operating temperature is 200° F or less, and

2. Maximum operating pressure is 275 psig or less.

NOTE: Piping which operates above the high-energy limits less than 2% of the time may be classified as moderate energy lines.

d. Normal Plant Conditions

Plant operating conditions normally experienced during reactor startup, operation at power, hot standby, or reactor cooldown to cold shutdown condition.

e. Upset Conditions

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

f. Postulated Piping Failures

Longitudinal and circumferential breaks in high energy fluid system piping and through-wall leakage cracks in high and moderate-energy fluid system piping postulated according to the provisions of Branch Technical Position (BTP) MEB 3-1, attached to Standard Review Plan (SRP) 3.6.2.

g. S_h and S_a

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

h. S_m

Design stress intensity as defined in Article NB-3600 of the ASME Code, Section III.

i. Single Active Component Failure

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

j. Terminal Ends

Extremities of piping runs that connect to structures, large components (e.g., vessels, pumps) or pipe anchors that act as rigid constraints to piping movement, including rotational movement, from static or dynamic loading. A branch connection to a main piping run is a terminal end of the branch run, except for cases where the piping model includes both the run and branch piping.

Intersections of runs of comparable size and stability are not considered terminal ends when the piping stress analysis model includes both the run and the branch piping, and the intersection is not rigidly constrained to the building structure.

k. Leakage Crack

A postulated opening in the piping system, the consequences of which are evaluated on the basis of pressure and temperature differential conditions and flooding effects.

l. Fluid Systems

High- and moderate-energy fluid systems that are subject to the postulation of piping failures against which protection of essential systems and components is needed.

3.6.1.1.2 Criteria

The pipe failure protection conforms to Appendix A of 10 CFR 50, General Design Criterion 4, Environmental and Missile Design Bases. The overall design for this protection is in compliance with NRC BTP APCSB 3-1 (attached to SRP 3.6.1), MEB 3-1 (attached to SRP 3.6.2) and NRC NUREG-1061, Volume 3 (Reference 12), the implementation of which is discussed herein.

3.6.1.1.3 Objectives

Protection against pipe failure effects was provided to fulfill the following objectives:

a. Assure that the reactor can be shut down safely and maintained in a safe shutdown condition or mitigate the consequences of a loss-of-coolant accident (LOCA).

b. Assure that containment integrity is maintained.

c. Assure that a pipe break does not directly or indirectly cause a loss of reactor coolant beyond makeup capability.

d. Assure that the radiological doses of a postulated piping failure will remain below the limits of 10 CFR 50.67.

3.6.1.1.4 <u>Assumptions</u>

The following assumptions were used to determine the protection requirements:

a. Pipe breaks or cracks were postulated to occur during normal plant operation (i.e., reactor startup, operation at power, hot standby or reactor cooldown to a cold shutdown).

b. Only high-energy piping as defined in Subsection 3.6.1.1.1.b and shown on Figure 3.6-1 was considered in the determination of the potential pipe break location. Moderate-energy piping as defined in Sub-section 3.6.1.1.1.c and shown on Figure 3.6-1 was capable of producing only cracks.

c. Pipe breaks were evaluated for the effects of pipe whip, jet impingement, flooding, room pressurization, and other environmental effects such as temperature.

d. Pipe cracks were evaluated for flooding and effects from spray.

e. Each longitudinal or circumferential break in high-energy fluid system piping, or leakage crack in moderate-energy fluid system piping, was considered separately as a single postulated initial event occurring during normal plant conditions. (Applicable to seismic and nonseismic piping).

f. Pipe failures (breaks or cracks) inside the containment were not evaluated concurrently with failures outside the containment.

g. Offsite power was assumed to be unavailable if a trip of the turbine-generator system or reactor protection system was a direct consequence of the postulated piping failure, unless it was more conservative to assume that offsite power was available (e.g., a feedwater line break with offsite power available leads to a larger inventory of water for flooding considerations).

h. A single active component failure was assumed in systems used to mitigate consequences of the postulated piping failure and to safely shut down the reactor, except as noted in paragraph i below. The single active component failure was assumed to occur in addition to the postulated piping failure, such as unit trip and loss of offsite power.

i. Where the postulated piping failure was assumed to occur in one of two or more redundant trains of a dual-purpose, moderate-energy essential system (i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping failure), single failures of components in the other train or trains of that system were not assumed provided the system was designed to the following criteria: (1) Seismic Category I standards: (2) has both offsite and onsite power sources; and (3) was constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety systems. Examples of systems that qualify as moderate-energy dual-purpose essential systems are the shutdown service water system and the residual heat removal system.

j. All available systems, including those actuated by operator actions, were employed to mitigate the consequences of a postulated piping failure to the extent clarified in the following paragraphs:

    1. In judging the availability of systems, account was taken of the postulated failure and its direct consequences such as unit trip and loss of offsite power, and of the assumed single active component failure and its direct consequences. The feasibility of carrying out operator actions was judged on the basis of ample time and adequate access to equipment being available for the proposed actions.

    2. Only Seismic Category I equipment could be used to mitigate the consequences of the failure and bring the plant to a safe shutdown.

k. A whipping pipe was not considered capable of rupturing impacted pipes of equal or greater nominal pipe diameter and equal or greater thickness.

l. Pipe movement was assumed to occur in the direction of the jet reaction.

m. Absorption of the fluid internal energy associated with the pipe break reaction could take into account any line restrictions (e.g., flow limiter) between the pressure source and break location and absence of energy reservoirs, as applicable.

n. Initial pipe break events are not assumed to occur in pump and valve bodies because of their greater wall thicknesses and their locations in the low stress portions of the piping systems.

o. Piping which is physically separated (or isolated) from structures, systems, or components important to safety by protective barriers, such as concrete incasements, will not initiate another pipe break event beyond the separated area.

3.6.1.1.5 Identification of Systems Important to Plant Safety

For a given postulated piping failure, the systems which may be required to shut down the plant and maintain it in a safe condition are identified in Table 3.6-1. Figures FP-8 thru FP-20 of Clinton Power Station Fire Protection Evaluation Report dated April 12, 1978, show the location on general arrangement drawings of all safety-related equipment including electrical cable trays.

Typical piping runs with postulated failure points indicated and the design approach to protect essential components are illustrated in Subsection 3.6.2.

High-energy fluid systems and divisional separation are shown in Figure 3.6-1 for each system important to plant safety.

Those portions of non-safety-related systems (i.e., service air, service water) which penetrate the primary containment were evaluated individually for divisional separation and postulated piping failure effects as described in Subsections 3.6.1.1.3 and 3.6.1.1.4.

3.6.1.2 Description

3.6.1.2.1 Potential Sources for Piping Failure

A list of systems considered high-energy per Subsection 3.6.1.1.1.b is given in Table 3.6-2. The boundaries for these high-energy, as well as for moderate-energy lines, are shown in Figure 3.6-1. The piping and instrumentation diagrams (P&ID's) in Figure 3.6-1 should not be used for detailed information, i.e. vents, drains. Detail information should be taken from P&ID's for each applicable system located throughout this USAR.

Locations, orientations, and size of piping failures within high-moderate-energy piping systems are postulated per the criteria given in Subsection 3.6.2.

The effects of pipe whip, jet impingement, spraying and flooding on essential systems, components, and equipment are discussed in Attachment D3.6, Failure Mode Analysis.

There are no credible secondary missiles formed from the postulated rupture of piping.

3.6.1.2.2 Structures and Compartments Used to Protect Against Piping Failure

Pressure response analyses were performed for the subcompartments containing high-energy piping. For a detailed discussion of the line breaks selected, vent paths, room volumes, analytical methods, pressure results, etc., refer to Subsection 6.2.1.2 for containment subcompartments and Table 3.6-3 for subcompartments located outside the containment.

Several subcompartments analyzed are also used for divisional separation and therefore contain only safety-related systems and components of one safety division. Subcompartments used as such were analyzed to determine the environmental and pressurization effects of pipe failure; but, in these cases, only system failure, not component failure, was analyzed in the Failure Mode Analysis (Attachment D3.6). Those subcompartments used for divisional separation are listed in Table 3.6-4 for the auxiliary building and the fuel building. (The circulating water screen house has no high-energy piping systems and therefore no subcompartment evaluations are included.)

3.6.1.2.3 Pipe Failure Effects on Control Room

There is no high-energy piping capable of producing impact or jet impingement effects in or near the control room. There are no effects upon the habitability of the control room by pipe break either from pipe whip, jet impingement, or transport of steam. Further discussion on control room habitability systems is provided in Section 6.4.

3.6.1.2.4 Impact of Plant Design for Postulated Piping Failures on Inservice Inspection

Access has been provided for inservice inspection as dictated by the ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Power Plant Components."

3.6.1.3 Safety Evaluation

3.6.1.3.1 General

In the plant design, consideration was given to the effects of postulated piping breaks with respect to the limits of acceptable damage/loss of function, to assure that, even with coincident single loss of active component, an earthquake equal to the safe shutdown earthquake, and loss of offsite power, the remaining structures, systems, and components would be adequate to safely shut down the plant. This subsection summarizes the Structural, Mechanical, Instrumentation, Electrical, and HVAC items that are safety-related and therefore designed to remain functional for the following: (1) a high-energy line rupture with resulting whip, impingement, compartment pressurization and temperature rise, wetting of compartment surfaces, and flooding, or (2) a moderate-energy break with resulting impingement, wetting of compartment surfaces, and flooding.

By means of the design features such as separation, barriers, and pipe whip restraints, all of which are discussed below, it has been assured that the design function or necessary component operability of essential items will not be impaired by the effects of postulated breaks and cracks.

Specific design features used for protecting the essential systems, components, and equipment are summarized in Attachment D3.6. The ability of specific safety-related systems to withstand a single active failure concurrent with a postulated event is discussed as applicable.

If a review of the pipe layout and plant arrangement drawings showed that the effects of the postulated breaks/cracks, on a reasonable basis, are isolated, physically remote, or restrained by plant design features from essential systems or components, no further evaluation was performed.

3.6.1.3.2 Protection Methods

a. General

The effects associated with a particular break/crack must be mechanistically consistent with the failure. Thus, actual pipe dimensions, piping layouts, material properties, and equipment arrangements were considered in defining the specific measures for protection against actual pipe movement and other associated consequences of postulated failures.

Protection against the dynamic effects of pipe failures was provided in the form of pipe whip restraints, equipment shields, and physical separation of piping, equipment, and instrumentation.

The precise method chosen depended largely upon limitations placed on the designer such as accessibility, maintenance, and proximity to other pipes and equipment.

b. Separation

The plant arrangement provides distance separation to the extent practicable between redundant safety systems (including their auxiliaries) in order to prevent loss of safety function as a result of the dynamic effects of pipe break or crack. Separation is the basic protective measure incorporated in the design to protect against dynamic effects.

c. Barriers, Shields, and Enclosures

Protection requirements were met with walls, floors, columns, and foundations in many cases.

d. Piping Restraints

Measures for protection against pipe whipping as a result of high-energy pipe breaks were not provided where any one of the following applied:

1. The piping was physically separated (or isolated) from all essential safety-related structure, systems, or components required to place the plant in a safe shutdown condition following the postulated rupture, or was restrained from whipping by plant design features such as concrete encasement.

2. Following a single break, the unrestrained movement of either end of the ruptured pipe could not damage, to an unacceptable level, any structure, system, or component required to place the plant in a safe shutdown condition following the postulated rupture.

3. The energy associated with the whipping pipe was demonstrated to be insufficient to impair, to an unacceptable level, the safety function of any structure, system, or component required to place the plant in a safe shutdown condition following the postulated rupture.

The design criteria for restraints are given in Subsection 3.6.2.

3.6.1.3.3 Specific Protection Measures

a. The general layout of the facility followed a multi-step process to ensure adequate separation.

1. Safety-related systems were located away from most high-energy piping.

2. Redundant (e.g., "A" and "B" trains) safety subsystems were located in separate compartments.

3. As necessary, specific components were enclosed to retain the redundancy required for those systems that must function as a consequence of specific piping failure.

4. Drainage systems were reviewed to assure their adequacy for flooding prevention.

b. For high-energy piping systems which penetrate the containment and drywell, isolation valves are located as close to the containment as possible to facilitate design of moment guides and maintain isolation valve operability.

c. The pressure, water level, and flow sensor instrumentation for those essential systems which are required to function following a pipe rupture are protected.

d. High-energy fluid system piping restraints and protective measures were designed such that a postulated break in one pipe could not, in turn, lead to a rupture of other nearby pipes or components if the secondary rupture could result in consequences considered unacceptable for the initial postulated break.

e. Deleted

f. For any postulated pipe rupture, the structural integrity of the containment was assured. In addition, for those postulated ruptures classified as a loss of reactor coolant, the design leak tightness of the containment fission product barrier was assured.

g. Safety/relief valves and the RCIC steamline were located and restrained so that a pipe failure would not prevent depressurization.

h. Separation was provided to preserve the independence of the low-pressure core spray (LPCS) and low-pressure coolant injection (LPCI) portions of the RHR systems.

i. High-energy piping which penetrated both the drywell and the containment was provided with guard pipes in accordance with Subsection 3.6.2.4. The encapsulated piping was designed in accordance with the criteria of Subsection 3.6.2.5.

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

Described herein are the design bases for postulating piping breaks and cracks inside and outside of containment, the procedures used to define the jet thrust reaction at the break location, the jet impingement loading criteria, and the piping dynamic response models.

3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration

3.6.2.1.1 Definition of High-Energy Fluid System

The definition of a high-energy fluid system is found in Subsection 3.6.1.1.1b.

3.6.2.1.2 Definition of Moderate-Energy Fluid System

The definition of a moderate-energy fluid system is found in Subsection 3.6.1.1.1c.

3.6.2.1.3 Postulated Pipe Breaks and Cracks

A postulated pipe break is defined as a sudden, gross failure of the pressure boundary either in the form of a complete circumferential severance (guillotine break) or as development of a sudden longitudinal split and is postulated for high-energy fluid systems only. For moderate-energy fluid systems, pipe failures are confined to postulation of controlled cracks in piping and branch runs. These cracks affect the surrounding environmental conditions only and do not result in whipping of the cracked pipe.

The following high-energy piping systems (or portions of systems) have been considered in the determination of a postulated pipe break during normal plant conditions and are evaluated for potential damage resulting from dynamic effects.

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.

c. All other piping systems or portions of piping systems considered high-energy systems.

A high-energy piping system break is not postulated as simultaneous with a moderate-energy piping system crack nor is any pipe break or crack outside containment postulated concurrently with a postulated pipe break inside containment.

A list of high-energy fluid systems is provided in Table 3.6-2.

3.6.2.1.4 Exemptions from Pipe Break Evaluation and Protection Requirements

The following are exemptions for postulating pipe breaks:

a. Piping is classified as moderate-energy piping.

b. The nominal pipe size is 1 inch or less.

c. The operation period is short. (An operation period is considered "short" if the fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is about 2 percent or less of the time that the system operates as a moderate-energy fluid system: e.g., systems such as the reactor decay heat removal system). Piping in this category is classified as moderate energy.

d. Portions of high-energy piping systems that are isolated from the source of the high-energy fluid during normal plant conditions are exempted from consideration of postulated pipe breaks. This would include portions of piping systems beyond a normally closed valve. This type of piping is classified as moderate-energy fluid system.

e. Pump and valve bodies are exempted from consideration of pipe break because of their greater wall thickness.

f. CRD insert lines are exempted per Reference 7.

Protection from pipe whip dynamic effects associated with pipe break is not provided if, following a single postulated pipe break, piping for which the unrestrained movement of either end of the ruptured pipe in any feasible direction about a plastic hinge, formed within the piping, cannot cause a loss of function of any structure, system, or component important to safety.

3.6.2.1.5 Types of Breaks and Leakage Cracks in Fluid System Piping

Except as noted in Subsection 3.6.2.1.4, the following types of breaks are postulated in high-energy fluid system piping:

a. Circumferential breaks are postulated only in piping exceeding a 1-inch nominal pipe diameter.

   Where break locations are selected in piping without the benefit of stress calculations, breaks are postulated concurrently at the piping welds to each fitting, valve, or welded attachment (not applicable to recirculation piping).

   Circumferential breaks are assumed to result in pipe severance separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections. Cases for which the resulting pipe separation is less than one pipe diameter because of physical limitation by piping restraints, structural members, or piping stiffness will be identified.

b. Longitudinal splits are postulated only in piping having a nominal diameter equal to or greater than 4 inches.

   Longitudinal breaks are assumed to result in an axial split without pipe severance. Splits are oriented (but not concurrently) at two diametrically opposed points on the piping circumference such that the jet reaction causes out-of-plane bending of the piping configuration. Alternatively, a single split is assumed at the section of highest tensile stress as determined by detailed stress analysis.

c. Circumferential breaks are assumed at all terminal ends and at intermediate locations identified by the criteria in Subsections 3.6.2.1.6.1 and 3.6.2.1.6.2.1.1. At each of the intermediate postulated break locations identified to exceed the stress and usage factor limits of the criteria in Subsections 3.6.2.1.6.1 and 3.6.2.1.6.2.1.1, either a circumferential or a longitudinal break, or both, are postulated per the following:

   1. Circumferential breaks are postulated at fitting joints.

   2. Longitudinal breaks are postulated in the center of the fitting at two diametrically opposed points (but not concurrently) located so that the reaction force is perpendicular to the plane of the piping and produces out-of-plane bending.

   3. Consideration is given to the occurrence of either a longitudinal or circumferential break. The state of stress in the vicinity of the postulated break location may be used to identify the most probable type of break. If the maximum stress range in the longitudinal direction is greater than

1.5 times the maximum stress range in the circumferential direction, only the circumferential break is postulated, and, consequently, if maximum stress range in the circumferential direction is greater than 1.5 the stress range in the longitudinal direction, only the longitudinal break is postulated. If there are no significant differences between the circumferential and longitudinal stresses, then both types of breaks are considered.

4. At intermediate locations chosen to satisfy the minimum break location criteria, only circumferential breaks are postulated (not applicable to recirculation piping).

d. For design purposes, a longitudinal break area is assumed to be the equivalent of one circumferential pipe area for all longitudinal breaks postulated in piping.

e. For both longitudinal and circumferential breaks, after assessing the contribution of upstream piping flexibilities, pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration for circumferential breaks and out-of-plane for longitudinal breaks and to cause pipe movement in the direction of the jet reaction.

f. For a circumferential break, the dynamic force of the jet discharge at the break location is based upon the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Thrust coefficients have been determined in accordance with ANS-58.2 (ANSI N176). Justifiable line restrictions, flow limiters, and the absence of energy reservoirs are used, as applicable, in the reduction of the jet discharge.

The following through-wall leakage cracks are postulated in high and moderate-energy fluid system piping at the locations specified in this position:

a. Cracks are postulated in moderate-energy fluid system piping and branch runs exceeding a nominal pipe size of 1 inch.

b. Fluid flow from a crack is based on a circular opening of area equal to that of a rectangle one-half pipe-diameter in length and one-half pipe wall thickness in width.

c. The flow from the crack is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments. Flooding effects are determined on the basis of a conservatively estimated time period required to effect corrective actions. Evaluation of jet impingement effects is not considered for postulated through-wall leakage cracks.

d. Through-wall leakage cracks instead of breaks are postulated in the piping of those fluid systems that qualify as high-energy fluid systems for only short operational periods as defined in Subsection 3.6.2.1.4.

3.6.2.1.6 Location for Postulated Pipe Breaks and Leakage Cracks

3.6.2.1.6.1 Criteria for Reactor Recirculation Piping System Inside Containment - Within the Scope of the NSSS Supplier

Postulate pipe break locations are selected in accordance with the U.S. Nuclear Regulatory Commission (NRC) Branch Technical Position APCSB 3-1, Appendix B NRC Branch Technical Position MEB 3-1, and NUREG-1061, Volume 3. For the ASME Section III, Class 1 recirculation piping system which is classified as high-energy, the postulated break locations are as follows:

a. At the terminal ends of the pressurized portions of the run.

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

    1. The maximum stress range between any two loads sets (including the zero load set) should not exceed 2.4 S_m and should be calculated by Eq. (10) in NB-3653, ASME Code Section III.

       If the calculated maximum stress range of Eq. (10) exceeds 2.4 S_m, the stress ranges calculated by both Eq. (12) and Eq. (13) in Paragraph NB-3653 should meet the limit of 2.4 S_m.

    2. The cumulative usage factor should be less than 0.1.

    3. The maximum stress, as calculated by Eq. (9) in NB-3652 under the loadings resulting from a postulated piping failure beyond these portions of piping should not exceed the lesser of 2.25 S_m and 1.8 S_y except that following a failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stresses provided a plastic hinge is not formed and operability of the valves with such stresses is assured in accordance with the requirements specified in SRP Section 3.9.3. Primary loads include those which are deflection limited by whip restraints.

Intermediate breaks are no longer postulated where the calculated cumulative usage factors and stress intensity ranges are lower than the limits specified in subparagraph b above (see Reference 11).

There are no Class 1 moderate energy piping systems.

There is no piping component in this subsection other than ASME Class 1.

3.6.2.1.6.2 Piping Other Than Reactor Recirculation Piping in Subsection 3.6.2.1.6.1

This subsection applies to all high- and moderate-energy piping inside and outside containment with the exception of the reactor recirculation piping in Subsection 3.6.2.1.6.1.

3.6.2.1.6.2.1 High-Energy Fluid System Piping

3.6.2.1.6.2.1.1 Fluid System Piping Not in the Containment Penetration Area

a. Breaks in ASME, Section III, Class 1 piping are postulated at the following locations in each piping run or branch run:

1. At terminal ends of the run;

2. At locations where the primary plus secondary stress intensity range between any two load sets (including the zero load set) as calculated by Equation (10) and either Equation (12) or (13) in paragraph NB-3653 of ASME, Section III exceeds 2.4 S_m for loadings resulting from normal and upset plant conditions, including SRV discharge and suppression pool vibratory loads and an OBE event; and

3. At any intermediate locations between terminal ends where the cumulative usage factor derived from the piping fatigue analysis under the loadings resulting from plant normal, upset, and testing conditions, SRV discharge and suppression pool vibratory loads, and an OBE event exceeds 0.1.

4. In the event that two intermediate locations cannot be determined by the stress or usage factor limits just described, the two locations of highest stress, as calculated by equation (10) in paragraph NB-3653 of ASME Section III, which are separated by a change in direction of the pipe run, are selected. If the piping run has only one change or no change of direction, only one intermediate break is postulated. A given elbow or other fitting (tee, reducer, etc.) is considered as a single-break location regardless of the number of types of breaks postulated at the fitting.

5. Breaks postulated using the criteria in item 4 are generally referred to as "arbitrary intermediate breaks" (AIBs). Initially AIBs were postulated for the following Class I piping (see Attachment B3.6 Figures):

MS0I	HP0I	RH05	NB01
MS02	LP01	RH34	MS05
MS03	RH01	RI01	RR32
MS04	RH03	SC07	RR33

However, recently, AIBs have been eliminated for this piping, since it is not susceptible to stress corrosion cracking, steam or water hammer effects, or thermal fatigue in fluid mixing situations (See Reference 11).

b. Breaks in ASME Section III Class 2 and 3 piping and seismically qualified ANSI B31-1 piping are postulated at the following locations-in each piping run or branch run:

1. At terminal ends of the run, and

2. At each intermediate location where the stresses under the loadings resulting from normal and upset plant conditions, SRV discharge and suppression pool vibratory loads (if applicable), and an OBE event, as calculated by the sum of equations (9) and (10) in paragraph NC-3652 of ASME Section III, exceed 0.8 (1.2 S_h + S_a).

3. In the event that two intermediate locations cannot be determined by the stress limits described above, the two locations of highest stress as calculated by the sum of equations (9) and (10) in paragraph NC-3652 of ASME Section III which are separated by a change in direction of the pipe run are selected. If the piping run has only one change or no change of direction, only one intermediate break is postulated. A given elbow or other fitting (tee, reducer, etc.) is considered as a single-break location regardless of the number of types of breaks postulated at the fitting.

4. Breaks postulated using the criteria in item 3 are referred to as "arbitrary intermediate breaks" (AIBs). Initially AIBs were postulated for the following Class 2 and 3 piping (see Attachment B3.6 Figures):

RT02	RT07	RH14
RT05	RT08	RH07
IS03	RI02	RH08
RT06		

   However, recently AIBs have been eliminated for this piping since it is not susceptible to stress corrosion cracking, steam or water hammer effects, or thermal fatigue in fluid mixing situations (see Reference 11).

5. As an alternative to the foregoing, intermediate locations are assumed at each location of potential high stress or fatigue such as pipe fittings, valves, flanges, and welded attachments.

c. Longitudinal breaks are not postulated at terminal ends.

d. Leakage cracks in high-energy ASME Section III Class 1 piping are postulated at locations where the primary and secondary stress intensity range, as calculated by equation (10) or equations (12) and (13) in paragraph NB-3653 of ASME Section III, exceed 1.2 Sm for loadings resulting from normal and upset plant conditions.

e. Leakage cracks in high-energy ASME Section III Class 2 and 3 piping and seismically qualified ANSI B31.1 piping are postulated at locations where the stresses under the loadings resulting from normal and upset plant conditions and an OBE event, as calculated by the sum of equations (9) and (10) in paragraph NC-3652 of ASME Section III, exceed 0.4 (1.2S_h + S_a).

f. Breaks and cracks in nonseismically qualified piping are postulated at locations to produce the controlling design basis events.

3.6.2.1.6.2.1.2 Fluid System Piping in Containment Penetration Areas

This subsection applies to the fluid system piping between the containment isolation valves, including the valves and the valve-to-process piping welds and any connection to the containment penetration.

Breaks are not postulated in the containment penetration area as defined above where the following design requirements are met.

a. The following design stress and fatigue limits are not exceeded for ASME Code Section III Class 1 piping:

   1. The primary plus secondary stress intensity range between any two load sets (including the zero load set) for normal and upset conditions, SRV discharge and suppression pool vibratory loads, and an OBE event, as calculated by equation (10) or equations (12) and (13), does not exceed 2.4 S_m.

   2. The cumulative usage factor U derived from the piping fatigue analysis under the loadings associated with normal, upset, and testing conditions, SRV discharge and suppression pool vibratory loads, and an OBE event is less than 0.1.

   3. The maximum stress, as calculated by equation (9) in paragraph NB-3652 under the loadings resulting from internal pressure, dead weight, and a postulated piping failure of fluid systems beyond these portions, does not exceed 2.25 S_m.

b. The following design stress and fatigue limits are not exceeded for ASME Code Section III Class 2 piping:

   1. The maximum stress range, as calculated by the sum of equations (9) and (10) in paragraph NC-3652, ASME Code Section III, under the loadings resulting from the normal and upset plant conditions (i.e., sustained loads, occasional loads including SRV discharge and suppression pool vibratory loads, and thermal expansion) and OBE event, does not exceed 0.8 (1.2 S_h +S_a).

   2. The maximum stress, as calculated by equation (9) in paragraph NC-3652 under the loadings resulting from internal pressure, dead weight, and a postulated piping failure of fluid system piping beyond these portions of piping does not exceed 1.8 S_h.

c. Following a piping failure outside the first pipe whip restraint, the formation of a plastic hinge is not permitted in the piping between the containment penetration and the first pipe whip restraint. Bending and torsion limiting restraints are installed, as necessary, at locations selected to optimize overall piping design, to prevent formation of a plastic hinge as noted, to protect against the impairment of the leaktight integrity of the containment, to assure isolation valve operability, and to meet the stress and fatigue limits in the containment penetration area.

d. Leakage cracks in the containment penetration area are postulated in accordance with Subsection 3.6.2.1.6.2.1.1.

e. The number of circumferential and longitudinal piping welds and branch connections is minimized as much as practical.

f. The length of these portions of piping is reduced to the minimum length practical.

g. An augmented ISI will be performed as discussed in Subsections 5.2.4.12 and 6.6.8.

The break exclusion areas are typically shown on the B3.6 Figures.

3.6.2.1.6.2.1.3 Details of the Containment Penetration

Details of the containment penetrations are discussed in Subsections 3.8.1 and 3.8.2.

3.6.2.1.6.2.2 Moderate-Energy Fluid System Piping Inside and Outside Containment

Leakage cracks in moderate-energy piping are postulated individually at locations that would result in the maximum effects from fluid spraying and flooding, with the consequent hazards or environmental conditions developed from the spray or flood. The consequences of these postulated breaks were analyzed for each room in the safety-related buildings. The results of these analyses are presented in USAR Section D.3.6.3.

3.6.2.1.7 Definitions

Throughout this section, applicable definitions are located in Subsection 3.6.1.1.1.

3.6.2.2 Analytical Methods to Define Forcing Functions and Response Models

3.6.2.2.1 Reactor Recirculation Loop Piping - Inside Containment

3.6.2.2.1.1 Analytical Methods to Define Blowdown Forcing Functions

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

a. Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one diameter lateral displacement of the ruptured piping sections.

   The details of the inelastic pipe whip analysis are provided in CPS-USAR Subsection 3.6.2.2.1.2, which describe methodol-gy of the GE computer program PDA used for this analysis.

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

ANS-58.2 (Reference 13) is the basic document that is used for determining the thrust coefficients in evaluating the dynamic force due to jet discharge. General Electric is using a value of 1.26 for a main steam line break and 2.0 for a recirculation line break. These values are conservative upper bound based on the following theoretical and experimental methods.

(1) <u>Saturated and Superheated Steam</u> (Main Steam Line)

For calculating the thrust force, saturated or superheated steam is treated as an ideal gas with a ratio of specific heat equal to 1.3. Considering the flow to be isentropic, the thrust coefficient, per Reference 14, for frictionless flow is given by:

$$C_T = 1.26 - P_a / P_o$$

where:

P_a = ambient pressure around pipe

P_o = pressure in the pipe

C_T = thrust coefficient

For the main steam high energy pipe, since $P_a \ll P_o$, $C_T \cong 1.26$

(2) <u>Sub-Cooled Water</u> (Recirculation Line)

As the degree of subcooling increases, the thrust coefficient for frictionless subcooled water increases from 1.26 in steam condition to a maximum of 2.0 for non-flashing water.

Normalization of the steady state thrust coefficient for frictionless flow of subcooled water based on the Henry-Fauske model (Reference 15) results in the following expression for C_T:

$$C_T = 2.0 - 0.861h^{*2};\ 0 \le h^* \le 0.75$$

$$C_T = 3.22 - 3.0h^{*} + 0.97h^{*2};\ 0.75 < h^* \le 1.0$$

where:

$h^* = (h_o - 180) / (h_{saturated} - 180)$

h_o = stagnation enthalpy (Btu/lbm)

$H_{saturated}$ = saturated water enthalpy at the stagnation pressure (Btu/lbm).

The experimental comparisons made by Hanson (Reference 16) showed good agreement with the theoretical prediction of thrust coefficient. In evaluating the dynamic force due to jet discharge, a conservative thrust coefficient of 2.0 is used for the recirculation piping. (MEB (DSER) 14)

c. Breaks are postulated to occur instantaneously (MEB (DSER) 14).

Blowdown forcing functions are determined by either of the two following methods:

a. The predicted blowdown forces on pipes fed by a pressure vessel are described by transient and steady-state forcing functions. The forcing functions used are based on methods described in Reference 3. These are simply described as follows:

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

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

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

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

$$F = \left[\left(P - P_a\right) + \frac{\rho \mu^2}{g_c} \right] A$$

where:

F = Blowdown Force

P = Pressure at exit plane

P_a = Ambient pressure

u = Velocity at exit plane

ρ = Density at exit plane

A = Area of break

g_c = Gravitational constant

5. Following the transient period a steady-state period is assumed to exist. Steady-state blowdown forces are calculated including frictional effects. For saturated steam, these effects reduce the blowdown forces from the theoretical maximum of 1.26 PoA. The method of accounting for these effects is presented in Reference 3. For subcooled water, a reduction from the theoretical maximum of 2.0 PoA is found through the use of Bernoulli's and standard equations such as Darcy's equation, which account for friction.

b. The following is an alternate method for calculating blowdown forcing functions:

The computer code RELAP3 (Reference 4) is used to obtain exit plane thermodynamic states for postulated ruptures. Specifically, RELAP3 supplies exit pressure, specific volume and mass rate. From these data the blowdown reaction load is calculated using the following relation:

$$\frac{T}{AE} = P_E - P\infty + \frac{G_E^{\;2}\,\bar{v}_E}{g_c}$$

$$R = \frac{-T}{AE} \quad x \quad AE$$

where:

T/AE	= thrust per unit break area	- lbf/ft^2,
P_E	= exit pressure	- lbf/ft^2,
$P\infty$	= receiver pressure	- lbf/ft^2,
G_E	= exit mass flux	- lb/sec ft^2,
$\bar{v}_E$	= exit specific volume	- ft^3/lbm,
g_c	= gravitational constant and	- 32.174 (ft-lb) / (lbf - sec^2),
R	= reaction force on the pipe	- lbf.

3.6.2.2.1.2 Pipe Whip Dynamic Response Analyses

The criteria used for performing the pipe whip dynamic response analyses include the following:

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

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

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

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

e. Piping within the broken loop is no longer considered part of the RCPB. Plastic deformation in the pipe is considered as a potential energy adsorber. The maximum strain in the pipe is limited to 25% of the ultimate uniform strain of the pipe material. This limit is the same as that imposed on the energy absorbing, plastically deforming pipe whip restraints. Piping systems are designed so that plastic instability does not occur in the pipe at the design dynamic and static loads unless damage studies are performed which show that direct damage to any essential system or component does not result.

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

The pipe whip analysis was performed using the PDA computer program (Reference 5). 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.

A comprehensive verification program has been performed to demonstrate the conservatisms inherent in the PDA pipe whip computer program and the analytical methods utilized. Part of this verification program included an independent analysis by Nuclear Services Corporation, under contract to the General Electric Company, of the recirculation piping system for the 1969 Standard Plant Design. The recirculation piping system was chosen for study because of its complex piping arrangement and assorted pipe sizes. The NSC analysis included elastic-plastic pipe properties, elastic-plastic restraint properties and gaps between the restraint and pipe and is documented in Reference 6. The piping/restraint system geometry and properties and fluid blowdown forces were the same in both analyses. However, a linear approximation was made by NSC for the restraint load-deflection curve supplied by GE. This approximation is demonstrated in Figure 3.6-2. The effect of this approximation is to give lower energy absorption of a given restraint deflection. Typically, this yields higher restraint deflections and lower restraint to structure loads than the GE analysis. The deflection limit used by NSC is the design deflection at one-half of the ultimate uniform strain for the GE restraint design. The restraint properties used for both analyses are provided in Table 3.6-5.

A comparison of the NSC analysis with the PDA analysis, as presented in Table 3.6-6, shows that PDA predicts higher loads in 15 of the 18 restraints analyzed. This is due to the NSC model including energy absorbing effects in secondary pipe elements and structural members. However, PDA predicts higher restraint deflections in 50% of the restraints. The higher deflections predicted by NSC for the lower loads are caused by the linear approximation used for the force - deflection curve rather than by differences in computer techniques. This comparison demonstrates that the simplified modeling system used in PDA is adequate for pipe rupture loading, restraint performance and pipe movement predictions within the meaningful design requirements for these low-probability postulated accidents.

3.6.2.2.2 Piping Other Than Reactor Recirculation Loop Piping - Inside Containment

This subsection applies to all high-energy piping, both inside and outside containment, excluding the piping considered to be part of the reactor recirculation loop.

3.6.2.2.2.1 Determination of Pipe Thrust and Jet Loads

3.6.2.2.2.1.1 Circumferential Breaks

The dynamic force of the jet discharge at the break location is based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically determined thrust coefficient. Line restriction flow limiters, positive pump controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of the jet discharge. Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration and to cause pipe movement in the direction of the jet reaction.

3.6.2.2.2.1.2 Longitudinal Breaks

The dynamic force of the fluid jet discharge is based on a circular break area equal to the cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of jet discharge. Piping movement is assumed to occur in the direction of the jet reaction unless limited by structural members, piping restraints, or piping stiffness.

3.6.2.2.2.1.3 Pipe Blowdown Force and Wave Force

The calculation of the magnitude and duration of the wave force acting on bounded pipe segments is based on a design guide for estimating discharge forces by Moody (Reference 1).

The calculation of the blowdown force is based on either an exact computer model (Reference 9) or on the following simplified conservative methodology.

The calculation of the blowdown force is consistent with Reference 1, and with Section 6.0 of ANSI N176 dated January, 1978 (Reference 2). If there is a fluid reservoir having sufficient capacity to develop a steady jet for a significant interval, the magnitude of the steady-state blowdown force used for saturated steam, saturated water, or a saturated steam and water mixture is equal to 1.26 P_oA_e for frictionless fluid flow (where P_o equals the stagnation pressure of the initial vessel fluid and A_e equals the break area). The magnitude of the steady-state blowdown force used for subcooled water varies from 1.26 P_oA_e to 2.0 P_oA_e for frictionless fluid flow depending on the degree of subcooling. However, the steady-state blowdown force is reduced by taking frictional effects into consideration as per Reference 2. For break locations where the frictional effects are significant, the blowdown force on the broken pipe segment is further reduced by considering the effect of wave propagation and reflection. Figure 3.6-3 shows the blowdown force on the pipe versus time for circumferential breaks. The pipe thrust used for longitudinal breaks is equal to the largest circumferential blowdown force at the same break location in accordance with Subsection 3.6.2.2.2.1.2. Nomenclature used in Figure 3.6-3 is defined below.

a. Three different blowdown magnitudes are calculated:

   1. $F_{impulse} = F_{imp} = P_oA_e$

   2. $F_{intermediate} = F_{int} = (P_oA_e - F_w)$

      $F_{imp} = F_{int}$ implies $F_w = 0$

      where

      F_w = wave force (transient),

      A_e = pipe flow area, and

      P_o = line pressure.

   3. $F_{steady\ state} = F_{ss}$

b. Fw initial is determined from Figure 9-23 of Reference 1. F_w initial for flashing water for pressures not shown in Figure 9-23 is equal to (P - 1.26 P_{sat}) A (where P_{sat} equals the saturation pressure of the initial pipe fluid).

c. $F_{steady\ state} = F_{ss}$ is determined in accordance with Reference 2.

d. T_{imp} = Time to $F_{intermediate}$ for circumferential breaks and is determined by dividing the distance to breaks and is determined the first elbow from the break by the sonic speed of the significant fluid wave. The sonic wave speed (C) is determined from Figure 9-29 of Reference 1.

e. F_{final} = The larger of F_{int} or F_{ss}.

3.6.2.2.2.2 Methods for the Dynamic Analysis of Pipe Whip

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). Simplified models of the local region near the break were analyzed to calculate displacement of the pipe and restraint. These calculated displacements were then used to calculate strains in the pipe, and were compared to allowable restraint deflection.

A finite difference model was used (Reference 10) for the pipe moment-curvature and the restraint resistance-displacement functions. The simplified models shown in Figure 3.6-5 were used to represent the local region near the break and to calculate the displacement in the restraint as well as the displacements and strains in the pipe.

3.6.2.2.2.2.1 Finite Difference Analysis

A finite difference formulation specialized to the case of a straight beam and neglecting axial inertia and large deflection effects is used for the analysis of pipe whip of stainless and carbon steel pipes. The dynamic analysis is performed by direct numerical time integration of the equations of motion.

The equations of motion are of the form:

$$h\ (P_k\ -\ m_k y''_k\ ) = -\ M_{k+1}\ +\ 2M_k\ -\ M_k\ -1 \qquad (3.6\text{-}15)$$

where:

h_k is the node spacing,

P_k is the externally applied lateral loads at node k,

m_k is the lumped mass at node k,

y_k is the lateral deflection at node k, and

M_k is the internal resisting moment in the beam at node k.

Power law moment-curvature relationship is assumed and the central difference approximation for the curvature

$$\frac{1}{h^2}\left(-y_{k+1} + 2y_k - y_{k-1}\right)$$

is used.

A timewise central-difference scheme is used to solve the dynamic equations

$$y_{(t+\Delta t)} = t^2 y''_{(t)} + y_{(t)} - y_{(t-\Delta t)} \qquad (3.6\text{-}16)$$

and for the first time step

$$y(\Delta t) = \Delta t^2 y_{(0)} \qquad (3.6\text{-}17)$$

A time step not more than 1/10 the shortest period of vibration is used in the integration.

3.6.2.2.2.2.1.1 Elastic-Plastic Moment Curvature Law

The pipe is assumed to obey an elastic-strain hardening plastic moment-curvature law with isotropic strain hardening. The symbols used are defined as follows:

M = moment,

$\overline{M}$ = current yield moment,

E = elastic modulus of material at temperature,

I = moment of inertia,

Z = EI,

ϕ = curvature,

ϕ_ε = M/z = elastic curvature,

$\Delta\phi\rho$ = increment of plastic curvature,

$\phi\rho$ = $\left|\sum\Delta\phi\right|$ = effective plastic curvature, and

ϕ_o = $\sum A\phi\rho$ = permanent set curvature.

At the end of each integration step, new values of o are calculated at each node.

The known values of $\phi\rho$, ϕ, and M at the start of the step are used to calculate M, $\overline{M}$, and $\Delta\phi\rho$ by the following procedure:

if $\left|\phi - \phi_o\right| < M/Z$,

M = z ($\phi - \phi_o$),

and $\Delta\phi\rho = 0$;

if $\left|\phi - \phi_o\right| > \overline{M}/Z$,

$$M = \overline{M} = F\left(\left|\phi - \phi_o\right| + \phi\rho\right) \sin\left(\phi - \phi_o\right),$$

and $\phi\rho = \phi - \phi_o - M / Z$,

where $F(\phi) = K(\phi)^n$

3.6.2.2.2.2.1.2 Power Law Moment Curvature Relationship

The following stress strain law is assumed in the plastic range:

$$\sigma = K\left(\frac{\sigma}{R_o}\right)^n \quad (3.6\text{-}18)$$

The corresponding moment curvature law is

$$M = K\,(\phi)^n \quad (3.6\text{-}19)$$

where:

$$K = \frac{2\sqrt{\pi}}{3+n}\left(R_o^{\,3+n} - R_i^{\,3+n}\right)\frac{r(n/2+1)\overline{K}}{r(n/2+3/2)} \quad (3.6\text{-}20)$$

or, to a good approximation,

$$K = \frac{4\overline{K}}{3+n}\left(1 - 0.291n - .076n^2\right)\left(R_o^{\,3+n} - R_i^{\,3+n}\right) \quad (3.6\text{-}21)$$

in which:

R_o = pipe outside radius, and

R_i = pipe inside radius.

In the elastic range, the moment-curvature law is:

$$M = EI\phi \quad (3.6\text{-}22)$$

The transition from elastic to plastic behavior on initial loading occurs at:

$$\phi = \frac{(EI)^{\frac{1}{n-1}}}{K} \quad (3.6\text{-}23)$$

3.6.2.2.2.2.1.3 Strain Rate Effects

The effect of strain rate in carbon steel is accounted for by using a rate dependent stress strain law of the form

$$\sigma(\varepsilon, \dot{\varepsilon}) = \left[\left(1 + \frac{\dot{\varepsilon}}{(40.4)} \right)^{1/5} \right] G(\varepsilon) \qquad (3.6\text{-}24)$$

where $G(\varepsilon)$ is the static stress strain relationship. For stainless steels, the effect of strain rate is less pronounced so that a 10% increase in yield and ultimate strengths is used. The selection of material properties is discussed in Attachment A3.6.

3.6.2.2.2.2.1.4 Restraint Behavior

The analysis is capable of handling the bilinear or power law restraint behavior as shown in Figure 3.6-7. The behavior of the restraint is unidirectional. The restraint unloads elastically only to zero state, being left with a permanent set, and reloads along the same curve as shown in Figure 3.6-7.

3.6.2.2.2.3 Method of Dynamic Analysis of Unrestrained Pipes

The impact velocity and kinetic energy of unrestrained pipes is calculated on the basis of the assumption that the segments on each side of the break act as rigid-plastic cantilever beams subject to piecewise constant blowdown forces. The hinge location is fixed either at the nearest restraint or at a point determined by the requirement that the shear at an interior plastic hinge is zero. The kinetic energy of an accelerating cantilever segment is equal to the difference between the work done by the blowdown force and that done on the plastic hinge. The impact velocity V is found from the expression for the kinetic energy:

$$KE = (1/2)\, M_{eq} V_I^2 \qquad (3.6\text{-}25)$$

where M_{eq} is the mass of the single degree of freedom dynamic model of the cantilever. The impacting mass is assumed equal to M_{eq}.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.3.1 Jet Impingement Analyses and Effects on Safety Related Components

3.6.2.3.1.1 Jet Impingement Criteria and Characteristics

The criteria used for evaluating the effects of fluid jets on safety related structures, systems and components are as follows:

a. Safety-related structures, systems and components are not impaired so as to preclude essential functions. For any given postulated pipe break and consequent jet, those structures, systems and components needed to safely shut down the plant are identified.

b. Safety related structures, systems and components which are not necessary to safely shut down the plant for a given break are not protected from the consequences of the fluid jet.

c. Safe shutdown of the plant due to postulated pipe ruptures of the reactor coolant pressure boundary (RCPB) is not jeopardized by sequential failures of safety-related piping. The required emergency core cooling system performance is maintained.

d. Offsite dose limits specified in 10 CFR 100 are complied with.

e. Postulated design-basis breaks resulting in jet impingement loads are assumed to occur in high-energy lines at full (100%) power operation of the plant.

f. Postulated through-wall leakage cracks are postulated in moderate-energy lines and are assumed to result in wetting and spraying of safety-related structures, systems and components.

g. Reflected jets are considered only when there is an obvious reflecting surface (such as a flat plate) which directs the jet onto a safety-related target. Only the first reflection is considered in evaluating potential targets.

h. Potential targets in the jet path are considered or the full extent of pipe displacement up to the calculated final position of the broken end of the ruptured pipe. This selection of potential targets is considered adequate due to the large number of breaks analyzed and the protection provided from the effects of these postulated breaks.

Jet impingement load calculations where the stagnation condition at the postulated break location is steam or saturated water between 60 to 170 BARS pressure (1 BAR = 14.7 psi), that were prepared after April 1, 1984 are based on NUREG/CR-2913 (Reference 8). NUREG/CR-2913 is a multidimensional computer study, which also accounts for the shock effects at the jet/target interface. When using the NUREG procedure, the impingement force includes the shape factor, $K\phi$, as defined in Reference 2.

Jet impingement load calculations outside the range of pressure where the NUREG procedure is applicable (less than 60 BARS), or calculations where the stagnation condition at the postulated break location is subcooled water, or calculations that were prepared before April 1, 1984, are based on the following simplified, one-dimensional procedure.

The analytical methods used to determine which targets are impingement upon by a fluid jet and the corresponding jet impingement load include:

a. The impinging jet proceeds along a straight path.

b. The total impingement force acting on any cross-sectional area of the jet is time and distance invariant, with a total magnitude equivalent to the fluid blowdown force as defined below.

c. The jet impingement force is uniformly distributed across the cross-sectional area of the jet, and only the portion intercepted by the target is considered.

d. The circumferential and longitudinal break opening is assumed to be a circular orifice of cross-sectional flow area equal to the effective flow area of the break.

e. The jet impingement force is equal to the steady state value of the fluid blowdown force as calculated by the methods described in Subsection 3.6.2.2.1.1.

f. The distance of jet travel is divided into two or three regions. Region 1 (see Figure 3.6-8) extends from the break to the asymptotic area. Within this region the discharging fluid flashes and undergoes expansion from the break area pressure to the atmospheric pressure. In Region 2 the jet remains at a constant diameter. In Region 3 interaction with the surrounding environment is assumed to start and the jet expands at a half angle of 10°.

g. Moody (Reference 1) has developed a simple analytical model for estimating the asymptotic area for steam, saturated water, and steam-water blowdown conditions.

   For fluids discharging from a break which are below the saturation temperature at the corresponding room pressure or have a pressure at the break area equal to the room pressure, free expansion does not occur. In these cases, the jet can be assumed to have a constant cross-sectional area equal to the break area.

h. For fluids which are above the saturation temperature at room pressure, the jet model expands at a half angle of 45° from the break to the asymptotic area (Region 1) for fully separated circumferential and longitudinal breaks. Assuming a linear expansion from the break area to the asymptotic area, the jet shape can be defined for Region 1 as well as Regions 2 and 3. Reference 2 is used to determine the asymptotic area.

i. Both longitudinal and fully separated circumferential breaks are treated similarly. The value of fl/D used in the blowdown calculation is also used for jet impingement.

j. Circumferential breaks with partial (i.e., l<D/2) separation between the two ends of the broken pipe, not significantly offset (i.e., no more than one pipe wall thickness lateral displacement) are more difficult to quantify; therefore, use of this procedure will be identified if applied. For these cases the following assumptions are made:

   1. The jet is uniformly distributed around the periphery.

   2. The jet cross section at any cut through the pipe axis has the configuration depicted in Figure 3.6-8(B) and the jet regions are as therein delineated.

   3. The jet force Fj = total blowdown F.

   4. The pressure at any point intersected by the jet is:

$$P_j = \frac{F_j}{A_R}$$

where:

A_R = the total 360° area of the jet at a radius equal to the distance from the pipe centerline to the target.

5. The pressure of the jet is then multiplied by the area of the target submerged within the jet in the manner explained in Paragraphs k and l.

6. The area (A_R) of the jet at target intersection distance r_t from pipe centerline is calculated by using Reference 2 to determine r_A (the distance from the pipe centerline to the plane of asymptoticity) and the relationship

$$A_R = 2\pi\, r_t\, l_R$$
$$[\text{See Figure 3.6-8 (B)}]$$

where:

A_A = asymptotic jet area

A_B = break area

D = pipe inside diameter

l = distance of pipe separation

l_A = width of jet at r_A and infinitely outward

l_R = width of jet at r_T

k. Target loads are determined using the following procedures and assumptions:

1. For both the fully separated circumferential breaks and the longitudinal breaks, the jet is assumed to reach its asymptoticity expanding at a half angle of 45° from the break. (Region 1, Figure 3.6-8(a)). For design purposes, the jet is assumed to have linear expansion within this region. The distance L from the break to the asymptotic area is calculated by

$$L = \frac{D_B}{2}\left[\left(\frac{A_A}{A_B}\right)^{1/2} - 1\right]$$

where:

A_A = asymptotic jet area

A_B = jet cross sectional area at break

D_B = diameter of jet at break area

The ratio of A_A/A_B is determined from Reference 2.

2. The area within Region 2 can be assumed to be constant out to the beginning of Region 3 which starts at the intersection of a line drawn at a 10° half angle dotted line Figure 3.6-8(a) and (c) and the boundary of the jet. In Region 3 the area expands at a constant 10° half angle.

3. After determination of the total area of the jet at the target, the jet pressure is calculated by

$$P_i = \frac{F_j}{A_x}$$

where:

P = incident pressure, and

A = area of the expanded jet at the target intersection.

4. The total force on any target which intercepts a portion of the jet is

$$F_{target} = K_o \; P_{jet} \; A_t$$

where:

A_t = the area of the target intercepted by the jet

K_o = the shape factor.

The shape factor is related to the drag coefficient, C_d, by $K_\phi = 1/2\ C_d$. Values of C_d are given in Reference 2.

l. For the partially separated circumferential breaks described in Paragraph j above, the target loads are calculated similarly, with the exception that the jet geometry is different according to Paragraph j and Figure 3.6-8(B).

Evaluation of the potential targets to withstand the jet impingement loads is performed.

For analysis of piping systems as targets, evaluation of design adequacy is based on the following load combination for the faulted condition:

Pressure + Weight + $(SSE^2 + Jet^2)^{1/2}$

Functional capability is evaluated when required.

3.6.2.3.1.2 Protective Measures

3.6.2.3.1.2.1 Protection and Analyses Guidelines

Protection against the dynamic effects of a pipe break is provided in the form of pipe whip restraints, equipment shields as required, 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 lines as described in Attachment B3.6 of the following systems inside the containment and dry well were analyzed for restraint against pipe whip and assessed for jet impingement:

   1. main steam
   2. feed water
   3. RHR
   4. RCIC
   5. LPCS
   6. HPCS
   7. RWCU
   8. reactor recirculation
   9. nuclear boiler
   10. standby liquid control

b. The lines as described in Attachment B3.6 of the following systems outside of the containment were assessed for jet impingement and analyzed against pipe whip:

   1. main steam
   2. feed water
   3. RCIC
   4. RWCU
   5. MSIV-LCS

Dynamic effects associated with the LOCA do not compromise the integrity of the containment and drywell.

The consequences of jet impingement do not result in any of the following:

a. inability to insert control rods,

b. inability to isolate the reactor coolant pressure boundary, and

c. inability to meet the core cooling system requirements.

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 able to perform their intended functions if loads are shown to be within allowable limits, otherwise, shields must be provided. Impacted passive equipment (pipes, restraints, and structures) are considered capable of continuing to perform their intended functions.

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.

3.6.2.3.1.2.2 Equipment Shields for Isolation

Equipment shields are selectively provided as required in order to isolate the equipment necessary to ensure segregation of the redundant systems of an accident and prevent it from causing a further chain accident. These shields are designed to with stand the rupture forces from piping and jets.

3.6.2.3.1.2.3 Jet Impingement Shields

Jet impingement shields are also selectively provided as required to limit the consequence of rupture of the piping and are designed to withstand the resultant jet forces.

3.6.2.3.1.2.4 Separation

Independence of redundant safety systems and components is maintained 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 its design function.

3.6.2.3.1.2.5 Acceptability of Analysis

The postulation of high energy line break locations and the conservative analysis of resulting jet thrust and impingement have been used to identify areas where restraints or other protection devices are required to protect safety-related systems and components.

3.6.2.3.2 Pipe Whip Effects on Safety-Related Components

This section of the USAR provides the criteria and methods used to evaluate the effects of pipe displacements on safety-related structures, systems and components following a postulated pipe rupture.

The criteria which are used for determining the effects of pipe displacements on components are as follows:

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

b. 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, if required, are met.

3.6.2.3.3 Pipe Whip Restraints

3.6.2.3.3.1 Functional Requirements

Pipe whip restraints differentiated from piping supports are designed to control the movement of a postulated ruptured pipe for an extremely low probability gross failure in a piping system carrying high-energy fluid. The piping integrity usually does not depend on the pipe whip restraints during normal, upset, emergency, or faulted conditions as defined in Section III of the ASME Boiler and Pressure Vessel Code. When piping integrity is lost because of a postulated break, the pipe whip restraint acts to limit the movement of the broken pipe to an acceptable distance.

The probability of pipe break accidents warrants that breaks be postulated in high-energy lines and that measures be taken to prevent consequential damage. The jet reaction force at the break is so large that snubbers and hangers not designed for pipe rupture loadings will usually be unable to prevent large displacement of the pipe. This large displacement of the pipe may cause damage to other mechanical, electrical, and structural systems necessary for safe shutdown of the plant. Also, if unrestrained, the blowdown thrust could produce strains equal to or greater than the ultimate strains in the pipe, resulting in local collapse of the pipe.

In order to mitigate the effects of pipe break, restraints are provided near the points of the postulated breaks. The restraints are designed to absorb the impact energy and to resist the steady-state blowdown force after absorbing the pipe whip energy. Pipe whip restraints allow free thermal movements at all times, during operation and shutdown of the plant.

3.6.2.3.3.2 Types of Pipe Whip Restraints

Three different types of pipe whip restraints are used as discussed below:

a. Tension Restraints

A typical tension restraint is shown in Figure 3.6-11. The tension restraint is composed of a variable assembly of U-shaped steel rods joined together to form a restraint of a specified resistance.

b. Crushable Material Restraints

Figure 3.6-12 shows a typical crushable material restraint. The crushable material absorbs energy only under compressive loading.

c. Two-legged Restraints

Figure 3.6-13 shows a typical two-legged restraint. Rods serve as yielding members for tensile loads. The energy associated with compressive loads is absorbed by the crushable energy-absorbing material.

3.6.2.3.3.3 Loading and Load Combinations

The pipe whip restraints are designed for the governing load combinations under abnormal/severe or abnormal/extreme loading conditions as per Table 3.8-2.

3.6.2.3.3.4 Design Requirements

For reactor recirculation piping, the dynamic analysis for pipe whip restraints is performed using the Pipe Dynamic Analysis (PDA) program as described in Subsection 3.6.2.2.1.2. For other piping, the dynamic analysis for pipe whip restraints is performed using the Pipe Whip Restraint Reaction Analysis (PWRRA) programs. This program provides resultant force-time histories which can then be input into the Response Spectrum Generation (RSG) program to generate dynamic load factors.

The yielding portion of the restraint is designed for the peak dynamic load. The non-yielding portion of the restraint is designed for the equivalent static load.

The functions of PWRRA are explained in detail in Subsection 3.6.2.2.2.2 and Appendix C, Section 25. The description of RSG is presented in Appendix C, Section 16.

3.6.2.3.3.5 Design Limits

Allowable steel stresses for non-yielding members are taken as 1.6 times AISC allowable but not more than 0.95 F_y where F_y = specified minimum yield stress.

Yielding in tension rods is limited to 50% of the ultimate strain.

Crushable material design is based on energy absorption principles. Deflection is controlled by the design energy. The honeycomb material thickness is designed such that the strain under this deflection is limited to 80% of the strain at which the honeycomb starts to strain-harden, and lies within the horizontal portion of the stress strain curve of the material. This ensures that the honeycomb material will not experience a deflection in excess of that defined by the horizontal portion of the load deflection curve.

3.6.2.4 Guard Pipe Assembly Design Criteria

The design of Code Class MC guard pipes and mechanical penetration assemblies conforms to ASME BPV Code, Section III, Division 1, 1974, including applicable addenda and Code Cases through Summer 1974. Details and design considerations of the Class MC guard pipe assemblies are discussed in Subsections 3.8.1.1.3.1, 3.8.1.5.3, and 3.8.1.5.5.

The inservice inspection of the guard pipe assemblies is in accordance with ASME BPV Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components (applicable edition/addenda as required by 10CFR50.55a). Penetration assembly components are arranged in a way that accessibility for periodic weld examinations and other inservice inspections, as applicable, are provided.

3.6.2.5 Material to be Submitted for the Operating License Review

3.6.2.5.1 Implementation of Criteria for Defining Pipe Break Location and Orientation

3.6.2.5.1.1 Postulated Pipe Breaks in Recirculation Piping System - Inside Containment

The criteria for selection of postulated pipe breaks in the recirculation piping system, inside containment, are provided in Subsection 3.6.2.1.6.1. The postulated pipe break locations and types selected in accordance with these criteria are shown in Figure B3.6-31. Conformance with these criteria is shown by Table B3.6-31.

3.6.2.5.1.2 Pipe Whip Restraints for Recirculation Piping System Inside Containment

The pipe whip restraints provided for this recirculation piping system are also shown in Figure B3.6-31. This system of restraints prevents unrestrained pipe whip resulting from a postulated rupture at any of the identified break locations.

3.6.2.5.1.3 Jet Effects for Postulated Ruptures of Recirculation Piping System - Inside Containment

The effects of jet impingement from breaks in the reactor recirculation piping are detailed in Subsection D3.6.2.4.

3.6.2.5.2 Piping Other Than Reactor Recirculation Piping

The following material pertains to the dynamic analyses applicable to piping systems inside and outside containment with the exception of the reactor recirculation loop piping.

3.6.2.5.2.1 Implementation of Criteria for Defining Pipe Break Locations and Configurations

The locations and number of design-basis breaks associated with whip restraints, including postulated rupture orientations for the high-energy piping systems, are based on the criteria delineated in Subsection 3.6.2.1 and are shown in Attachment B3.6.

3.6.2.5.2.2 Implementation of Criteria Dealing With Special Features

Special protective devices in the form of pipe whip restraints and impingement shields are designed in accordance with Subsection 3.6.2.3.

Pipe whip restraint locations, configurations, and orientations in relation to break locations are included in Attachment B3.6.

Where special protective devices are located in the vicinity of welds requiring augmented inservice inspection, one or both of the following criteria are met:

a. Special protective devices are located at such a distance from all welds so as to allow inservice inspection.

b. Special protective devices are removable so that inservice inspection can be performed.

3.6.2.5.2.3 Acceptability of Analyses Results

The postulation of break locations for high energy piping systems and analyses of the resulting jet thrust, impingement and pipe whip effects have been considered.

Postulated pipe break results are included in Attachment B3.6.

3.6.2.5.2.4 Design Adequacy of Systems, Components, and Component Supports

For each of the postulated breaks, the equipment and systems necessary to mitigate the consequences of the break and to safely shut down the plant (i.e., all essential systems and components) are identified in Subsection 3.6.1. The equipment and systems are protected against the consequences of each of the postulated breaks and cracks to ensure that their design-intended functions will not be impaired to unacceptable levels.

Where it is necessary to restrict the motion of a pipe that would result from a postulated break, pipe whip restraints are included in the respective piping systems, or structural barriers or walls are designed to prevent the whipping of the pipe.

Design adequacy of the restraints is included in Attachment B3.6. Results of a typical restraint analyses are given in Table B3.6-35

The structure and structural barriers are designed to withstand the effects of jet impingement loads. The loading combinations and allowable design limits are given in Tables 3.8-1.1, 3.8-1.2, and 3.8-2.

The evaluation of essential components under dynamic effects associated with jet impingement is presented in Attachment D3.6.

3.6.2.5.2.5 Implementation of Criteria Related to Protective Assembly Design

Guard pipes are discussed in Subsection 3.6.2.4.

3.6.3 References

1. P. T, Lahey, Jr. and F. J. Moody, "Pipe Thrust and Jet Loads," The Thermal-Hydraulics of a Boiling Water Nuclear Reactor, Section 9.2.3. pp. 375-409, Published by American Nuclear Society, Prepared for the Division of Technical Information, United States Energy Research and Development Administration, 1977.

2. ANSI N176 Design Basis for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Rupture, Draft, January 1978

3. GE Spec. No. 22A2625 - "System Criteria and Applications for Protection Against the Dynamic Effects of Pipe Breaks."

4. RELAP3 - A computer Program for Reactor Blowdown Analysis IN-1321, issued June 1979, Reactor Technology TID-4500.

5. GE Report NEDE-10313 - "PDA - Pipe Dynamic Analysis Program for Pipe Rupture Movement" (Proprietary Filing)

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

7. GE Safety Evaluation Report for the Design of GESSAR-238, NSSS (Docket No. STN50-550), page 3-4.

8. NUREG/CR-2913, SAND 82-1935, R4, "Two-Phase Jet Loads."

9. RELAP4/MOD5, Computer Program User's Manual, 09.8.026-5.5.

10. Pipe Whip Restraint Reaction Analysis User's Manual, 09.5.125-2.1.

11. NUREG-0853, "Safety Evaluation Report Related to the Operation of Clinton Power Station, Unit No. 1, "Supplement No. 5, January 1986.

12. NUREG-1061, Volume 3, November 1984.

13. ANS-58.2 (ANSI N176), "Proposed American National Standard Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture."

14. Shapiro, A.H., "The Dynamics and Thermodynamics of Compressible Fluid Flow", Vol, Ronald Press, NY, 1965.

15. Webb, S.W., "Evaluation of Subcooled Water Thrust Forces", Nuclear Technology, Vol. 31, October 1976.

16. Hanson, G.H., "Subcooled - Blowdown Forces on Reactor System Components: Calculation Method and Experimental Configuration", Idaho Nuclear Corporation Report IN-1354, June 1970.

TABLE 3.6-1
SYSTEMS IMPORTANT TO PLANT SAFETY

Piping Systems

Main Steam (including the Automatic Depressurization System) (MS)

Feedwater (FW)

Reactor Recirculation (RR)

Diesel Oil (DO) Fuel Pool Cooling & Cleanup (FC)

Shutdown Service Water (SX)

Suppression Pool Makeup (SM)

MSIV Leakage Control (IS)

Low Pressure Core Spray (LP)

High Pressure Core Spray (HP)

Residual Heat Removal (including the Low Pressure Coolant Injection) (RH)

Control Rod Drive (RD)

Standby Liquid Control (SC)

Reactor Core Isolation Cooling (RI)

Control Room HVAC (VC)

Diesel Generator Room Ventilation (VD)

Standby Gas Treatment (VG)

Shutdown Service Water Ventilation (VH)

Essential Switchgear Heat Removal (VX)

ECCS Pump Room Cooling (VY)

Refrigeration Piping Switchgear Heat Removal (RG)

Other Systems

Neutron Monitoring

Containment Atmosphere Monitoring

Leak Detection

Process Radiation Monitoring

Safety Related Standby Power (i.e., Diesel Generator, Auxiliary DC Power System)

Containment Isolation Systems

TABLE 3.6-2
HIGH-ENERGY FLUID SYSTEMS

SYSTEMS	NOTES
Main Steam (MS)	
Extraction Steam (ES)	(1)
Feedwater (FW)	
Condensate (CD)	(1)
Condensate Booster (CB)	(1)
Control Room HVAC (VC)	
Heater Drains (HD)	(1)
Misc. Vents & Drains (DV)	(1)
Turbine Drains (TD)	(1)
Turbine Gland Steam Seal Steam (GS)	(1)
MSIV Leakage Control (IS)	
Reactor Recirculation (RR)	(2)
Low Pressure Core Spray (LP)	(2)
High Pressure Core Spray (HP)	(2)
Nuclear Boiler (NB)	(2)
Residual Heat Removal (RH)	(2)
Reactor Water Cleanup (RT)	
Standby Liquid Control (SC)	(2)
Reactor Core Isolation Cooling (RI)	
Control Rod Drive (RD)	(2)
Off Gas (OG)	(1)
Radwaste Chemical Waste Process (WF)	(1)
Chemical Radwaste Reprocessing & Disposal (WZ)	(1)
Radwaste Sludge Process (WX)	(1)
Auxiliary Steam (AS)	(1)
Post Accident Sampling (PS)	(2)
Containment Monitoring (CM)	(2)
Laboratory HVAC (VL)	

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(1) Not considered an initiating system for piping failure because of complete physical separation from safety related systems, components and structures. (These systems are located in Non-Category I structures where there are no safety related components or systems.)

(2) The only high-energy portions of these systems are those portions which make up the reactor coolant boundary. (See Figure 3.6-1 for exact boundaries.)

TABLE 3.6-3
SUMMARY OF SUBCOMPARTMENT PRESSURIZATION ANALYSES

CUBICLE DESIGNATION	SUBCOMPARTMENT DESCRIPTION	VOLUME (ft^3)	HEIGHT (ft)	FLOW CROSS SECTIONAL AREA (ft^2)	BOTTOM ELEVATION (ft)	INITIAL CONDITIONS TEMP (°F)	INITIAL CONDITIONS PRESS. (psia)	INITIAL CONDITIONS HUMID. (%)	DBA BREAK CONDITIONS BREAK LOC. VOL. NO.	DBA BREAK CONDITIONS BREAK LINE	DBA BREAK CONDITIONS BREAK AREA (ft^2)	BREAK TYPE*	CALC. PEAK PRESS. DIFF. (psig)	DESIGN PEAK PRESS. DIFF. (psig)
1	Aux. Bldg. Floor Drain Sys.	14,650.0	22.50	250.0	712.00	122.0	14.7	0.1	1	1RH04A14	0.0091	L	1.25	1.25
2	RHR Pump Rm. 'C'	33,650.0	28.50	700.0	707.50	122.0	14.7	0.1	2	1RH08A14	0.0091	L	0.60	0.60
3	RHR Pump Rm. 'B'	24,650.0	26.75	610.0	707.50	104.0	14.7	0.1	4	1RH40AB10	0.7882	C	5.07	5.07
4	RHR Heat Exchanger Rm. 'B'	50,500.0	91.00	555.0	707.50	104.0	14.7	0.1	4	1RH40AB10	0.7882	C	5.54	5.54
5	RCIC Pump and Turbine Rm.	26,000.0	26.50	750.0	707.50	108.0	14.7	0.1	5	1RI05A4	0.1660	C	6.16	6.16
6	RHR Heat Exchanger Rm. 'A'	49,100.0	91.00	535.0	707.50	104.0	14.7	0.1	6	1RH40AA10	0.7882	C	5.54	5.54
7	RHR Pump Rm. 'A'	23,700.0	26.75	450.0	707.50	104.0	14.7	0.1	6	1RH40AA10	0.7882	C	5.07	5.07
8	LPCS Pump Rm.	48,892.0	28.25	660.0	707.50	104.0	14.7	0.1	8	1RH03AA14	0.0091	L	0.90	0.90
9	Corridor	82,900.0	28.25	465.0	707.50	104.0	14.7	0.1	14	1RT02AB3	0.0751	C	0.34	0.34
10	Ground Floor West	104,000.0	23.50	1100.0	737.00	104.0	14.7	0.1	10	1RH04A14	0.0091	L	0.48	0.48
11	MSIV Rms.	9,486.0	13.00	200.0	737.00	150.0	14.7	0.1	5	1RI05A4	0.1660	C	2.34	2.34
12	Ground Floor East	59,700.0	23.50	1100.0	737.00	104.0	14.7	0.1	14	1RT02AB3	0.0751	C	0.40	0.40
13	Personnel Access Area	10,500.0	10.50	165.0	737.00	104.0	14.7	0.1	13	1RH03AA14	0.0091	L	0.60	0.60
14	RWCU Pump Rms.	1,275.0	10.00	35.0	737.00	104.0	14.7	0.1	14	1RT02AB3	0.0751	C	4.7	4.7
15	Pipe Chase	485.0	9.00	14.0	750.00	122.0	14.7	0.1	15	1RT02F6	0.3300	C	10.95	10.95
16	Pipe Tunnel	14,000.0	8.50	250.0	750.50	122.0	14.7	0.1	16	1RT02F6	0.3300	C	3.84	3.84
17	Mezzanine Floor West	78,700.0	17.50	785.0	762.00	104.0	14.7	0.1	10	1RH04A14	0.0091	L	0.18	0.18
18	Mezzanine Floor East	78,400.0	17.50	785.0	762.00	104.0	14.7	0.1	18	1RH03AA14	0.0091	L	0.48	0.48
19	Main Floor West	80,950.0	17.75	810.0	781.00	104.0	14.7	0.1	18	1RH03AA14	0.0091	L	0.36	0.36
20	Main Floor East	86,650.0	19.00	865.0	781.00	104.0	14.7	0.1	18	1RH03AA14	0.0091	L	0.36	0.36
A	Auxiliary/Main Steam Tunnel	141,356	Varies	Varies	Varies	150	14.7	0.1	A	1MS01EA24 1FW02EA20	3.1300 9.318	C	13.8	13.8

*L = 'through-wall leakage crack'
C = 'circumferemtial pipe rupture'

TABLE 3.6-4
SUBCOMPARTMENTS USED FOR DIVISIONAL SEPARATION

SUBCOMPARTMENT	LOCATION
RHR Pump "A" Cubicle (and RHR Heat Exchanger "A" Cubicle)	El. 707 ft-6 in., Auxiliary Building
RHR Pump "B" Cubicle (and RHR Heat Exchanger "B" Cubicle)	El. 707 ft-6 in., Auxiliary Building
RHR Pump "C" Cubicle	El. 707 ft-6 in., Auxiliary Building
LPCS Pump Cubicle	El. 707 ft-6 in., Auxiliary Building
RCIC Pump Cubicle	El. 707 ft-6 in., Auxiliary Building
HPCS Pump Cubicle	El. 707 ft-6 in., Fuel Building
Safety-Related Division 1 Switchgear Area	El. 781 ft-0 in., Auxiliary Building (Area bounded by column row 117 wall on the west, containment and column row AD walls on the south, column row 124 wall on the east, and column row S wall on the north.)
Safety-Related Division 2 Switchgear Area	El. 781 ft-0 in., Auxiliary Building (Area bounded by column row 102 wall on the west, column row AD and containment walls on the south, column row 107 wall on the east, and column row S wall on the north.)

TABLE 3.6-5
RESTRAINT DATA

General Restraint Data for 1 Bar of a Restraint
$F = C_2 (\Delta \text{ restraint})^n$
Where Δ restraint = δ pipe - total clearance

Pipe Size (In)	Rest Load Direction	C_2	n	Limit Restraint	Initial Clearance	Effective Clearance	Total Clearance
12	0°	27,733	-24	6.129	4	1.941	5.941
12	90°	14,795	-401	9.063	4	12.247	16.247
16	0°	109,265	-24	6.278	4	1.934	5.934
16	90°	62,599	-377	8.978	4	12.187	16.187
24	0°	102,228	-24	8.222	4	1.984	5.984
24	90°	55,531	-375	11.972	4	13.685	17.685
24	38°*	109,888	-24	5.588	4	5.698	9.698
24	52°*	109,835	-24	5.473	4	8.462	12.462

* Applies to Restraint RCR 3 only.

TABLE 3.6-6
COMPARISON OF PDA AND NSC CODE

Break ID No.*	Restraint ID No.*	Force Vector (degrees)(a)	No. Bars		Load (kips)		Restraint Deflection (in.)		Percent of Design Restraint Deflection		Pipe Deflection (in.)	
			PDA	NSC	PDA	NSC	PDA	NSC	PDA	NSC	PDA	NSC
RC1J	RCR1	0	5	5	803.3	788.3	6.6	7.9	79.9	96.4	17.7	15.6
RC2LL	RCR1	90	5	5	766.4	458.4	15.0	7.5	125.2	62.6	35.8	24.5
RC3LL	RCR2	0	6	6	747.0	639.7	2.3	3.7	27.7	45.4	17.2	20.1
RC3LL	RCR2	90	6	6	796.6	780.3	10.2	10.5	85.4	88.1	41.5	43.0
RC4LL	RCR3	0	5	5	846.0	838.4	8.2	8.1	99.2	98.0	18.9	16.4
RC4LL	RCR3	52	8	8	1,319.0	1,073.9	5.4	4.2	99.2	76.9	23.4	17.3
RC4CV	RCR3	38	8	8	1,260.7	1,275.0	4.5	5.6	80.4	99.9	22.6	18.7
RC6AV	RCR3	38	8	8	928.5	722.5	1.3	1.8	22.5	31.7	23.7	95.4
RC7J	RCR7	0	6	6	953.3	801.6	6.3	5.8	76.4	70.1	16.5	21.6
RC8LL	RCR6	90	4	4	599.0	NA**	8.3	NA	69.2	NA	26.8	NA
RC8LL	RCR7	90	6	6	895.0	NA	8.2	NA	68.2	NA	29.3	NA
RC9CV	RCR6	0	4	4	575.8	520.2	4.2	5.5	50.6	67.3	13.2	14.6
RC9LL	RCR8	90	6	6	830.2	546.8	11.4	6.8	95.3	56.9	36.7	26.2
RC11A	RCR8	90	6	6	818.3	493.6	11.0	6.0	91.7	50.1	31.4	23.7
RC12	RCR9	0	6	6	NA	832.9	NA	6.3	NA	76.9	NA	15.7
RC13	RCR10	0	4	4	668.4	478.0	5.9	3.7	93.5	58.4	13.4	10.4
RC14CV	RCR20	0	8	8	285.0	309.6	2.8	5.9	46.3	95.9	15.5	14.0
RC14LL	RCR20	90	8	8	116.3	129.9	1.0	3.4	10.5	37.1	22.0	23.6
RC16	RCR11	0	4	4	687.4	518.4	6.6	4.4	105.1	69.9	15.4	10.2

* See Figure 3.6-2

** NA – Data Not Available

(a) Force Vector Represented as -

TABLE 3.6-7
MASS AND ENERGY RELEASE RATE DATA
LINE BREAK IN STEAM TUNNEL
OUTSIDE CONTAINMENT

TIME (sec)	LIQUID MASS FLOW RATE (lb_m/sec)		STEAM MASS FLOW RATE (lb_m/sec)		LIQUID ENTHALPY (BTU/lb_m)		STEAM ENTHALPY (BTU/lb_m)		TOTAL MASS RELEASE RATE (lb_m/sec)	TOTAL ENERGY RELEASE RATE (BTU/sec)
	FWL*	MSL*	FWL	MSL	FWL	MSL	FWL	MSL		
0	24041	0	0	10600	408.5	550	0	1189.9	34641	2.24E + 07
0.028	24041	0	0	10600	408.5	550	0	1189.9	34641	2.24E + 07
0.029	18603.5	0	0	10600	408.5	550	0	1189.9	29203.5	2.02E + 07
0.34	18603.5	0	0	10600	408.5	550	0	1189.9	29203.5	2.02E + 07
0.341	5307.5	0	0	10600	408.5	550	0	1189.9	15907.5	1.48E + 07
1.000	5307.5	0	0	10600	408.5	550	0	1189.9	15907.5	1.48E + 07
1.001	5307.5	17395	0	4605	408.5	550	0	1189.9	27307.5	1.72E + 07
1.750	5307.5	17395	0	4605	408.5	550	0	1189.9	27307.5	1.72E + 07
1.751	5307.5	20400	0	1600	408.5	550	0	1189.9	27307.5	1.53E + 07
4.000	5307.5	20400	0	1600	408.5	550	0	1189.9	27307.5	1.53E + 07
5.500	5307.5	0	0	0	408.5	550	0	1189.9	5307.5	2.17E + 06
6.000	5307.5	0	0	0	408.5	550	0	1189.9	5307.5	2.17E + 06

The feedwater line break mass and energy releases were also evaluated at a reduced feedwater temperature (RFWT) of 380°F and an enthalpy of 354.8 Btu/lbm. The subcompartment analyses were also completed for the case of reduced feedwater temperature (RFWT) at LPU conditions. The resulting feedwater line break mass release increased by 5% and the energy releases increased by 3%.

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* FWL - Feedwater Line
MSL - Main Steam Line

TABLE 3.6-8
SUBCOMPARTMENT NODAL DESCRIPTION
LINE BREAK IN STEAM TUNNEL
OUTSIDE CONTAINMENT

		INITIAL CONDITIONS					DBA BREAK CONDITIONS				
VOLUME NO.	DESCRIPTION	CROSS-SECTIONAL AREA (ft^2)	VOLUME (ft^3)	TEMP. (°F)	PRESS. (psia)	HUMID. (%)	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA (ft^2)	BREAK TYPE	CALC.** PEAK PRESS DIFF. (psig)
1	Main Steam Tunnel	562	67549	150	14.7	.1	CASE 1*				13.8
2	Main Steam Tunnel	590	44948	150	14.7	.1	CASE 2*				8.2
3	Main Steam Tunnel	1100	28859	150	14.7	.1	CASE 3*				6.0
4	Turbine Building: Basement, Grade Flr. Mezzanine Flr.	2000	780300	104	14.7	.1					-
5	Turbine Building: Main Flr.	3000	275000	104	14.7	.1					-
6	Atmosphere	10000	10^{10}	104	14.7	.1					-

* Indicates the break location for each case. The break was a simultaneous double-ended guillotine break of one main steam line and one feedwater line.

** The calculated peak pressure difference = the peak node pressure -14.7 psia.

TABLE 3.6-9
SUBCOMPARTMENT VENT PATH DESCRIPTION
LINE BREAK IN STEAM TUNNEL
OUTSIDE CONTAINMENT

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW CHOKED UNCHOKED	AREA (ft^2)	LENGTH (ft)	INERTIA (ft^{-1})	HYDRAULIC DIAMETER (ft)	FRICTION K, ft/d	TURNING LOSS, K	EXPANSION, K	CONTRACTION, K	TOTAL
1*	1	2	UNCHOKED	406.5	31.7	.078	22.8					2.5
2	2	3	UNCHOKED	590.0	40.1	.068	27.4					1.6
3	3	4	UNCHOKED	635.0	8.3	.013	28.4					2.1
4**	4	5	UNCHOKED	976.5	10.7	.011	35.3					4.1
5***	5	6	UNCHOKED	2000.0	16.0	.008	50.5					2.9
6****	0	1		1.0	00.0	.000	00.0					0.0

* Opened on a differential pressure in either direction of 0.37 psid.

** Opened on a differential pressure of 0.4 psid from 4 to 5.

*** Opened on a differential pressure of 0.7 psid from 5 to 6.

**** The break flow: CASE 1 into Node 1, CASE 2 into Node 2, CASE 3 into Node 3.

ATTACHMENT A3.6

SELECTION OF PIPE MATERIAL PROPERTIES FOR USE IN PIPE WHIP ANALYSIS

ATTACHMENT A3.6
SELECTION OF PIPE MATERIAL PROPERTIES
FOR USE IN PIPE WHIP ANALYSIS

The selection of yield and ultimate strength values for piping for use in pipe whip analysis is discussed in Section 6.3.2.3 of Reference 1. This part of the standard permits the use of representative or actual test data values of material properties. Minimum ASME code values can also be used with more conservative results. A substantial amount of elevated temperature test data for A106 Grade B carbon steel is given in Reference 2. Material property values based on this data are obtained and used.

Since little test data is available for TP304, TP304L, and TP316 stainless steels, ASME code specified values are used with the realization that they are very conservative.

The power law stress strain relationship is used for all steels.

$$s = K\varepsilon^{n} \qquad (1)$$

The effect of strain rate in carbon steels is accounted for (as suggested in Reference 3) by modifying Equation (1) as follows:

$$\sigma(\varepsilon, \dot{\varepsilon}) = \left[1 + \left(\frac{\dot{\varepsilon}}{D}\right)^{1/p}\right] K(\varepsilon)^{n} \qquad (2)$$

D = 40.4 sec

p = 5

This modification has been widely used (see for examples References 3 and 4). For stainless steels, the effect of strain rate is less pronounced (Reference 5) so that the use of a 10% increase in yield and ultimate strengths as suggested in Reference 1 is used.

<u>A106 Grade B Material Properties at 600° F</u>

The results of tests on 73 specimens are given in Reference 2. Twenty-one were tested at 600° F, twenty-two at room temperature, and the rest at temperatures between 200° F and 585° F. Yield stress (0.2% offset values were measured) was shown to decrease, ultimate stress to increase, with increasing temperature. The minimum yield stress of any of the 73 specimens tested was 31.6 ksi, and the average for the 21 tested at 600° F was 36.01 ksi with a sigma value of 3.6 ksi. The minimum ultimate stress value for all specimens was 64.4 ksi, and the average for the 22 tested at room temperature 71.79 ksi with a sigma value of 4.72 ksi. The strength coefficient K and the hardening exponent <u>n</u> can be evaluated from the following equations:

$$\sigma y = K \ (.002) \qquad (3)$$

$$\sigma u = Kn^{n} \qquad (4)$$

Values of K and n obtained in this way are given in the following tabulation:

	Yield Stress σy (ksi)	Ultimate Stress σu (ksi)	K (ksi)	n
Minimum	31.60	64.40	86.486	0.16201
Mean-Sigma	32.41	67.07	90.277	0.16484
Mean	36.01	71.79	96.080	0.15792

(Material properties in the tabulation, for A106 Grade B at 600° F, are based on data from Reference 2.)

The mean-sigma values of K = 90.277 ksi and n = 0.16484 are used for all temperatures 600° F and below.

References

1. ANSI N176 Design Bases for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Rupture, Draft, January 1978.

2. R. J. Eiber, et. al. Investigation of the Initiation and Extent of Ductile Pipe Rupture, Battelle Memorial Institute, Report BMI-1866, July, 1969.

3. S. R. Bodner and P. S. Symonds, "Experimental and Theoretical Investigation of the Plastic Deformation of Cantilever Beams Subjected to Impulsive Loading," JAM, December, 1962.

4. J. C. Anderson and A. K. Singh, "Inelastic Response of Nuclear Piping Subjected to Rupture Forces," ASME Paper No. 75-PVP-21.

5. C. Albertini and M. Montagnani, "Wave Propagation Effects in Dynamic Loadings," Nuclear Engineering and Design, 37-115-124, 1976.

ATTACHMENT B3.6

POSTULATED PIPE BREAK RESULTS

ATTACHMENT B3.6
POSTULATED PIPE BREAK RESULTS

Attachment B3.6 presents specific details discussed in Subsection 3.6.2.5.

The data presented is described below:

1. Location of all break locations that are postulated using the stress criteria or that require whip restraints are shown in Figures B3.6-1 through B3.6-34. These figures also show all required whip restraints.

2. The type of pipe break postulated, the pipe stresses, and the allowable stresses at the postulated break locations i are shown in Tables B3.6-1 through B3.6-34. Typically, the allowable stresses are lower bound values based on a temperature that envelopes all piping locations in the subsystem. These allowables are conservative for pipe rupture analysis purposes, and do not necessarily correspond to the allowable stresses used in the code stress analysis of these subsystems.

3. Typical results of pipe whip restraint analyses inside containment for high pressure core spray system are identified in Table B3.6-35.

4. Typical results to demonstrate design adequacy of those portions of high-energy piping penetrating containment for which additional stress criteria apply (i.e., break exclusion piping), and for which valve operability requirements must be met, are shown in Table B3.6-36.

TABLE B3.6-1
BREAK DATA, FEEDWATER SUBSYSTEM FW-01
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
FW-C21	C	42480	58165	8160	43491	0.086
FW-C31	C	42480		Terminal End Break		
FW-C32	C,L	42480	47075	28735	23042	0.211
FW-C33	C,L	42480	60181	47159	14654	0.312
FW-C34	C,L	42480	69546	56638	13573	0.435
FW-C40	C	42480		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-2
BREAK DATA, FEEDWATER SUBSYSTEM FW-02
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
FW-C1	C	42480	58165	8160	43491	0.086
FW-C11	C	42480		Terminal End Break		
FW-C12	C,L	42480	47075	28735	23042	0.211
FW-C13	C,L	42480	60181	47159	14654	0.312
FW-C14	C,L	42480	69546	56638	13573	0.435
FW-C20	C	42480		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-3
BREAK DATA, FEEDWATER SUBSYSTEM FW-03
OUTSIDE CONTAINMENT

BREAK** NUMBER	TYPE*	ALLOWABLE STRESS (psi) $0.8(1.2S_h + S_A)$	CALCULATED STRESS (psi) (EQ.9B + EQ.10)
3	C	Not Applicable	Not Applicable
4	L	Not Applicable	Not Applicable
5	C	Not Applicable	Not Applicable
6	C	Not Applicable	Not Applicable
7	L	Not Applicable	Not Applicable
8	C	Not Applicable	Not Applicable
9	C	Not Applicable	Not Applicable
A3	C	Not Applicable	Not Applicable
A4	L	Not Applicable	Not Applicable
A5	C	Not Applicable	Not Applicable
A6	C	Not Applicable	Not Applicable
A7	L	Not Applicable	Not Applicable
A8	C	Not Applicable	Not Applicable
A9	C	Not Applicable	Not Applicable
FW03-1	C	Not Applicable	Terminal End
FW03-2	C	Not Applicable	Terminal End

* Break type: C = circumferential, L = longitudinal

**Breaks were based on fitting criteria.

TABLE B3.6-4
BREAK DATA, HPCS SUBSYSTEM HP-01
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
HP-C9	C	42480		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-5
BREAK DATA, LPCS SUBSYSTEM LP-01
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
LP-C9A	C	42480		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-6
BREAK DATA, MAIN STEAM SUBSYSTEM MS-01
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
MS-C68	L,C	42480	67675	49264	18661	0.0423
MS-C69	C	42480		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-7
BREAK DATA, MAIN STEAM SUBSYSTEM MS-02
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
MS-C33	L,C	42480	64551	47389	23406	0.0306
MS-C34	C	42480		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-8
BREAK DATA, MAIN STEAM SUBSYSTEM MS-03
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
MS-C56	C	42480		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-9
BREAK DATA, MAIN STEAM SUBSYSTEM MS-04
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
MS-C20	C	42480		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-10
BREAK DATA, MAIN STEAM DRAIN SUBSYSTEM MS-05
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
MS-C69	C	42480		Terminal End Break		
MS-C78	C	42480		Terminal End Break		
MS-C87	C	42480		Terminal End Break		
MS-C93	C	42480	39658	--	--	.130
MS-C94	C	42480		Terminal End Break		

* Break type: C = circumferential..

TABLE B3.6-11
BREAK DATA, MAIN STEAM SUBSYSTEM MS-06
OUTSIDE CONTAINMENT

BREAK** NUMBER	TYPE*	ALLOWABLE STRESS (psi) $0.8(1.2S_h + S_A)$	CALCULATED STRESS (psi) (EQ.9B + EQ.10)
MS-C201	C	Not Applicable	Not Applicable
MS-C202	C	Not Applicable	Not Applicable

* Break type: C = circumferential.

**Breaks were based on fitting criteria.

TABLE B3.6-12
BREAK DATA, MAIN STEAM SUBSYSTEM MS-07
OUTSIDE CONTAINMENT

BREAK** NUMBER	TYPE*	ALLOWABLE STRESS (psi) $0.8(1.2S_h + S_A)$	CALCULATED STRESS (psi) (EQ.9B + EQ.10)
3L	L	Not Applicable	Not Applicable
4C	C	Not Applicable	Not Applicable
5L	L	Not Applicable	Not Applicable
6C	C	Not Applicable	Not Applicable
9L	L	Not Applicable	Not Applicable
10C	C	Not Applicable	Not Applicable
11L	L	Not Applicable	Not Applicable
12C	C	Not Applicable	Not Applicable
A3L	L	Not Applicable	Not Applicable
A4C	C	Not Applicable	Not Applicable
A5L	L	Not Applicable	Not Applicable
A6C	C	Not Applicable	Not Applicable
A9L	L	Not Applicable	Not Applicable
A10C	C	Not Applicable	Not Applicable
A11L	L	Not Applicable	Not Applicable
A12C	C	Not Applicable	Not Applicable

* Break type: C = circumferential, L = longitudinal

**Breaks were based on fitting criteria.

TABLE B3.6-13
BREAK DATA, MAIN STEAM DRAIN SUBSYSTEM MS-38A
OUTSIDE CONTAINMENT

All High Energy Piping in Subsystem MS-38A is Break Exclusion.

There are no postulated breaks on this subsystem.

TABLE B3.6-14
BREAK DATA, RHR SUBSYSTEM RH-01
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
RH-C10	C	42480		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-15
BREAK DATA, RHR SUBSYSTEM RH-03
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
RH-C26	C	42480		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-16
BREAK DATA, RHR SUBSYSTEM RH-05
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
RH-C18	C	42480		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-17
BREAK DATA, RHR SUBSYSTEM RH-34
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
RH-C35	C	47580		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-18
BREAK DATA, RHR SUBSYSTEM RH-07
OUTSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	ALLOWABLE STRESS (psi) 0.8 ($1.2S_h + S_A$)	CALCULATED STRESS (psi) (EQ.9B + RELIEF VIV + EQ.10B)

* Break type: C = circumferential, L = longitudinal.
Footnote: Terminal End Break at Penetration 1AB-0204 is not postulated as explained in Section D3.6.4.

TABLE B3.6-19
BREAK DATA, RHR SUBSYSTEM RH-08
OUTSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	ALLOWABLE STRESS (psi) 0.8 ($1.2S_h + S_A$)	CALCULATED STRESS (psi) (EQ.9B + RELIEF VIV + EQ.10B)

* Break type: C = circumferential, L = longitudinal.
Footnote: Terminal End Break at Penetration 1AB-0202 is not postulated as explained in Section D3.6.4.

TABLE B3.6-20
BREAK DATA, RCIC SUBSYSTEM RI-01
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
RI-C11	C	42240		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-21
BREAK DATA, RCIC SUBSYSTEM RI-02/RH-14
OUTSIDE CONTAINMENT

BREAK** NUMBER	BREAK** TYPE*	ALLOWABLE STRESS (psi) 0.8 ($1.2S_h + S_A$)	CALCULATED STRESS (psi) (EQ.9B + EQ.10)
RH-CA5	C	32400	Terminal End Break
RH-CA55	C	32400	Terminal End Break

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-22 HAS BEEN DELETED.

TABLE B3.6-23
BREAK DATA, RWCU SUBSYSTEM, RT-01
INSIDE CONTAINMENT

Note: See Calculation 066204(EMD) "Fatigue Analysis of Piping Subsystem 1RT-01" for Stress and Usage Factors for RWCU Subsystem, RT-01.

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	EQ. 12	EQ. 13	CUMULATIVE USAGE FACTOR
RT-C1	C			TERMINAL END BREAK		
RT-C28	C,L					
RT-C28B	C			TERMINAL END BREAK		
RT-C28C	C					
RT-C35A	C			TERMINAL END BREAK		
RT-C39	C					
RT-C40	C					
RT-C40A	C					
RT-C58A	C,L					
RT-C27B	C,L					
RT-C39A	C					
RT-C79	C			TERMINAL END BREAK		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-24
BREAK DATA, RWCU SUBSYSTEM RT-02
INSIDE CONTAINMENT

BREAK** NUMBER	TYPE*	ALLOWABLE STRESS (psi) 0.8 ($1.2S_h + S_A$)	CALCULATED STRESS (psi) (EQ.9B + EQ.10)
RT-C151A	C	32400	Terminal End Break
RT-C161A	C	32400	Terminal End Break
RT-C251A	C	32400	Terminal End Break
RT-C255A	C	32400	Terminal End Break
RT-C266A	C	32400	Terminal End Break
RT-C315A	C	32400	Terminal End Break
RT-C365A	C	32400	Terminal End Break

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-25
BREAK DATA, RWCU SUBSYSTEM RT-05
INSIDE CONTAINMENT

BREAK NUMBER	TYPE*	ALLOWABLE STRESS (psi) 0.8 ($1.2S_h + S_A$)	CALCULATED STRESS (psi) (EQ.9B + EQ.10)
RT-C173B	C	32400	Terminal End Break
RT-C174C	C	32400	Terminal End Break
RT-C181	C	32400	Terminal End Break
RT-C191	C	32400	Terminal End Break
RT-C203A	C	32400	Terminal End Break
RT-C211A	C	32400	Terminal End Break
RT-C221A	C	32400	Terminal End Break
RT-C203B	C	32400	46700
RT-C203C	C	32400	34700

* Break type: C = circumferential, L = Longitudinal.

TABLE B3.6-26
BREAK DATA, RWCU SUBSYSTEM RT-06
OUTSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	ALLOWABLE STRESS (psi) 0.8 ($1.2S_h + S_A$)	CALCULATED STRESS (psi) (EQ. 9B + EQ. 10)
RT-603	C	32400	Terminal End Break
RT-604	C	32400	Terminal End Break
RT-605	C	32400	Terminal End Break

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-27
BREAK DATA, RWCU SUBSYSTEM RT-07
OUTSIDE CONTAINMENT

BREAK NUMBER	TYPE*	ALLOWABLE STRESS (psi) 0.8 ($1.2S_h + S_A$)	CALCULATED STRESS (psi) (EQ. 9B + EQ. 10)
RT-C703	C	32400	Terminal End Break
RT-C704	C	32400	Terminal End Break
RT-C705	C	32400	Terminal End Break

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-28
BREAK DATA, RWCU SUBSYSTEM RT-08
OUTSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	ALLOWABLE STRESS (psi) 0.8 (1.2S_h + S_A)	CALCULATED STRESS (psi) (EQ. 9B + EQ. 10)
RT-C801	C	32400	Terminal End Break
RT-C803	C	32400	Terminal End Break
RT-C805	C	32400	Terminal End Break

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-29
BREAK DATA, RWCU DRAIN SUBSYSTEM RR-32
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
RT-C85	C	32940		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-30
BREAK DATA, RWCU DRAIN SUBSYSTEM RR-33
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
RT-C95	C	32940		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-31
BREAK DATA, REACTOR RECIRCULATION (RR)
INSIDE CONTAINMENT*

BREAK IDENT.	STRESS RATIO PER ASME EQNS. EQ(10) $\frac{S}{2.4\ S_m}$	EQ(12) $\frac{S}{2.4\ S_m}$	EQ(13) $\frac{S}{2.4\ S_m}$	USAGE FACTOR	BREAK TYPE***	BREAK BASES SECTION NO.
RS1	0.654	0.230	0.464	0.0	C	3.6.2.1.6.1.a
RS3** LL	1.676	0.911	0.619	0.30	L	3.6.2.1.6.1.b
RD1	0.620	0.351	0.409	0.0	C	3.6.2.1.6.1.a
RD2	0.796	0.185	0.374	0.0	C	3.6.2.1.6.1.a
RD3	0.701	0.094	0.394	0.0	C	3.6.2.1.6.1.a
RD4	0.650	0.279	0.411	0.0	C	3.6.2.1.6.1.a
RD5	0.747	0.506	0.376	0.0	C	3.6.2.1.6.1.a

* Loop A same as Loop B except as noted.
** Loop B only. Subscript "LL" indicates longitudinal break.
***Break type: C - circumferential, L - longitudinal

TABLE B3.6-32
BREAK DATA, MSIV-LEAKAGE CONTROL SUBSYSTEM IS-03
OUTSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	ALLOWABLE STRESS (psi) 0.8 $(1.2S_h + S_A)$	CALCULATED STRESS (psi) (EQ. 9B + EQ. 10)
1S-C1	C	32400	Terminal End Break
1S-C2	C	32400	Terminal End Break
1S-C3	C	32400	Terminal End Break
1S-C4	C	32400	Terminal End Break

* Break type: C = circumferential, L = Longitudinal.

TABLE B3.6-33
BREAK DATA, SLCS SUBSYSTEM SC-07
OUTSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
SC-C2	C	40080		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-34
BREAK DATA, NUCLEAR BOILER SUBSYSTEM NB-01
INSIDE CONTAINMENT

BREAK NUMBER	BREAK TYPE*	$2.4S_m$ (psi)	CALCULATED STRESS (psi) EQ.10	CALCULATED STRESS (psi) EQ. 12	CALCULATED STRESS (psi) EQ. 13	CUMULATIVE USAGE FACTOR
NB-C1	C	42480		Terminal End Break		
NB-C20	C	42480		Terminal End Break		

* Break type: C = circumferential, L = longitudinal.

TABLE B3.6-35
RESULTS OF WHIP RESTRAINT ANALYSIS FOR
HIGH PRESSURE CORE SPRAY INSIDE CONTAINMENT

PIPING SYSTEM						RESTRAINT INFORMATION*			
POSTULATED BREAK ID	RESTRAINT ID	F_{imp} (kips)	T_{imp} (10^{-3} sec.)	F_{FINAL} (kips)	GAP (inches)	TIP DIS-PLACEMENT inches)	ACTUAL DEFLEC-TION (INCHES)	PEAK DYNAMIC LOAD (kips)	ALLOWABLE DEFLECTION (inches)
HP-C9:C	HP-R5	-	-	100.95	7.3	19.25	4.42	278.5	6.29

* Restraint information is based on current analysis.

TABLE B3.6-36
RESULTS OF CONTAINMENT PENETRATION PIPING ANALYSIS
FOR FEEDWATER INSIDE CONTAINMENTNote:

			STRESS (psi)	
BREAK NUMBERS	RESTRAINT NUMBER (GUIDE)	PEAK RESTRAINT REACTION (kips)	MAXIMUM PIPE STRESS IN CONTAIN-MENT PENE-TRATION AREA	ALLOWABLE
FW-C32(C)	FW-R16	270		
&			23845	44381
FW-C32(L)	FW-R16A	462		

Note: Mirror image of feedwater line has the same stress and load results

TABLE B3.6-37
BREAK DATA, FWLCS, SUBSYSTEMS RH-85 AND RH-86
OUTSIDE CONTAINMENT

RH-85

BREAK NUMBER	BREAK TYPE*	ALLOWABLE STRESS (psi) $0.8(1.2S_h + S_a)$	CALCULATED STRESS (psi) (EQ. 9B + EQ. 10)
RH-85-1	C	N/A	Terminal End
RH-85-2	C	N/A	Terminal End

RH-86

NO BREAKS POSTULATED AS THIS PORTION OF HIGH ENERGY PIPING IS IN THE BREAK EXCLUSION AREA.

* BREAK TYPE: C = circumferential, L = longitudinal

ATTACHMENT C3.6
EVALUATION OF ESSENTIAL COMPONENTS
UNDER DYNAMIC EFFECTS OF JET IMPINGEMENT

This attachment has been deleted. The evaluation of essential components under the effects of jet impingement is covered in Section 3.6, which discusses jet forces and geometries. Attachment B3.6 lists the break locations in high energy piping. Attachment D3.6 discusses piping layout.

ATTACHMENT D3.6

<u>SUMMARY OF FAILURE MODE ANALYSIS FOR PIPE BREAKS AND CRACKS</u>

ATTACHMENT D3.6
SUMMARY
OF
FAILURE MODE ANALYSIS FOR PIPE BREAKS AND CRACKS

D3.6.1 GENERAL

This Attachment describes the specific pipe failure protection provided to satisfy the requirements of Subsection 3.6.1 and demonstrates that essential systems, components and equipment are not adversely affected by pipe breaks or cracks.

The information is divided into three categories: (1) a discussion of high-energy pipe breaks and the dynamic effects of pipe whip and jet impingement (Subsection D3.6.2); (2) a discussion of moderate-energy pipe cracks and the effects of spraying or wetting (Subsection D3.6.3); and (3) a discussion of flooding as a result of breaks or cracks (Subsection D3.6.4).

The primary method used throughout the plant to protect essential systems, components and equipment was physical separation. In order to effect physical separation of safety systems, certain generalized procedures were followed during the design stages of the project. All piping, mechanical equipment, electrical components and instrumentation for each of the safety systems in each of the divisions (3 Divisions of Engineered Safety Features and 4 Divisions of Electrical Components) were numbered to indicate the division in which they belong. In the design of the systems and in the layout of the general arrangements, all equipment and components in each of the safety divisions were marked with different colors so that it was easily determined from looking at drawings that they have been given separation from equipment and components in the other divisions. This technique facilitated the location of systems not in the safety division which need be checked for their impact on safety systems. Improper interconnections between safety divisions also were located by this mechanism. For those safety-related systems located outside the containment, physical separation was the primary mode of protection in all but a few isolated cases. These cases are described in the following subsections.

In general, the following design techniques applicable to the equipment in each safety division were applied throughout the plant:

a. The equipment within a single safety division was maintained together physically throughout the plant.

b. The distance to be maintained between equipment within different safety divisions precluded disabling or degrading of more than one nuclear safety-related division from a single event.

c. Non-nuclear safety-related equipment which contains high-energy pipelines or presents the potential for plant flooding (see Section 3.4) was located to preclude the disabling or degrading of more than one nuclear safety-related division by a single event.

d. In areas where adequate distance could not be maintained between two or more safety-related divisions or between high-energy non-divisional safety-related equipment and a nuclear safety related division, a failure mode analysis approach was taken to determine that the safety-related equipment involved was not adversely affected by pipe breaks or cracks in the immediate area. If the analysis indicated that a nuclear safety related system required protection, barriers, i.e., cubicles, walls, and tunnels were employed (see e. below).

e. In areas where cubicles or wall-type barriers are impractical, restraints were employed between high energy lines and the nuclear safety-related equipment.

The implementation of separation barriers was as follows: (1) distance separation; and (2) general or area barriers, such as rooms and walls (see Figures 3.5-3 through 3.5-5, Missile-Proof Walls).

Color-coded piping and instrument diagrams and color-coded composite diagrams were used to ensure that the routing of high-energy lines throughout the plant did not adversely affect essential systems, components and equipment.

It should be noted here that the pipe whip analysis was performed for high-energy lines as discussed in Section D3.6.2.

Figures D3.6-1 through D3.6-132 show only the high-energy piping (cross-hatched) and divisional piping, equipment, ductwork and instrument lines. Two inch and smaller piping and instrument tubing or electrical conduit which is presently being routed will be protected in accordance with the requirements of Subsection 3.6.1.

For a further description of pipe/crack locations and types, break exclusion areas (no-break zones, that is, areas where a pipe break is not postulated due to meeting the criteria of the Standard Review Plan (SRP) and Branch Technical Position (BTP) MEB 3-1, guard pipes, and pipe whip restraints, refer to Subsection 3.6.2. Attachment B3.6 identifies all pipe whip restraints and associated break locations on isometric drawings.

High-energy lines 8-inch nominal diameter and larger in the areas of the containment drywell are restrained; therefore, the dynamic effects of pipe whip are minimal on essential components. The most limiting problem in these areas is jet impingement. To protect from jet impingement, essential components were separated from other divisions and from high energy lines.

D3.6.2 <u>HIGH-ENERGY PIPING</u>

High-energy fluid systems are considered to be pipe rupture initiating systems. These systems are listed in Table 3.6-2.

Several of the high-energy systems are located in areas or buildings which house no safety-related systems, equipment or components. Therefore, these high-energy systems cannot impact on safety-related equipment and were not analyzed. Those systems are as follows:

a. extraction steam,

b. condensate,

c. condensate booster,

d. heater drains,

e. miscellaneous vents and drains,

f. turbine drains,

g. turbine gland steam seal steam,

h. radwaste chemical waste process,

i. chemical radwaste reprocessing and disposal,

j. radwaste sludge process, and

k. auxiliary steam.

The remaining high-energy fluid systems are described in the following subsections. Appropriate isometric drawings with break locations and restraints are shown in Attachment B3.6 and Figures D3.6-1 through D3.6-133. These figures depict areas both inside and outside the containment.

For this analysis, the movement of the restrained piping (tip displacement) was calculated using the PWRRA program as described in Subsection 3.6.2.3.3.

D3.6.2.1 Main Steam Piping

The location of the postulated pipe breaks and the pipe whip restraints for the main steam system is shown on Figures B3.6-6 through B3.6-13. The stress analysis used for the main steam system is summarized in Tables B3.6-6 through B3.6-13.

D3.6.2.1.1 General

Each of the four 24-inch main steamlines is welded to the appropriate reactor nozzle at elevation 797 feet-1/2 inch. This is approximately 7 feet above the top of the shield wall. After an elbow, the pipe is routed downward to elevation 771 feet, then horizontally around the reactor to the area between azimuthal angles 341° and 19°, where all four main steamlines then pass through their respective inboard MSIV's and the drywell wall penetrations. The main steamlines then pass through the containment steam tunnel, inside guardpipes, exiting the north wall of the containment and entering the auxiliary building main steam tunnel. Within the auxiliary building main steam tunnel, the main steamline passes through the out board MSIV, a third safety-related isolation valve, and runs horizontally through the steam tunnel into the turbine building.

No breaks were postulated from the inboard main steam isolation valve and extending outside containment beyond the third isolation valve through the north wall of the auxiliary building steam tunnel up to the first elbow fitting from the isolation valve (Figures B3.6-6 through B3.6-13). The piping from the reactor vessel through the second isolation valve satisfies all the requirements of ASME Code, Section III, Class 1, Quality Group A. Between the second and third isolation valves, the piping is Class 2, Quality Group B. After the third isolation valve, the piping complies with ANSI Standard B31.1, Quality Group D.

A total of 16 safety/relief valves are mounted on the horizontal runs between the reactor and the first isolation valve inside the drywell. The discharge piping and vent lines from these safety/relief valves are normally unpressurized; therefore, there is no potential for dynamic pipe whip or similar hazards.

In addition, an 8-inch line branching from main steamline A supplies steam to the RCIC turbine. This line, which passes through the containment and auxiliary building steam tunnels, is discussed in the analysis of the RCIC System in Subsection D3.6.2.10.

D3.6.2.1.2 Inside Drywell

The dynamic load of a nonrestrained whipping main steam pipe could impact several systems: the 8-inch RCIC, the 12-inch LPCI, the 10-inch LPCS, the 10-inch HPCS, the CRD hydraulic system, and the ADS relief valve system. A ruptured main steamline would rapidly depressurize the reactor as discussed in Chapters 6 and 15, therefore, the RCIC, the HPCS and the ADS systems would be unnecessary in mitigating the consequences of a main steamline break.

To preclude any likelihood of loss of a system required for safe plant shutdown, pipe restraints and guides have been installed inside the drywell as shown on the isometric and the composite drawings (see Attachment B3.6 Figures).

The environmental conditions are the same as the local environment in the drywell. All Class 1E electrical equipment in the drywell has been qualified (refer to Section 3.11 for environmental qualification).

The effects of jet impingement from breaks in the main steam piping have been evaluated. In the event of a postulated main steam line break, any equipment hit by the break and required for safe shutdown has sufficient redundant equipment not hit by the jet, or is sufficiently separated from the break so that the equipment can withstand jet force.

D3.6.2.1.3 Inside the Containment Steam Tunnel

The main steam piping from the deflection-limiting restraint inboard of the inboard isolation valve, through the steam tunnel in the containment, has been qualified as a no-break zone. This piping is inside guard pipes, consequently, no analysis of failure modes was performed.

D3.6.2.1.4 Inside the Auxiliary Building Steam Tunnel

The main steam piping in the auxiliary building steam tunnel has no breaks from the containment wall and extending outside containment beyond the third isolation valve and through the north wall of the auxiliary building steam tunnel into the turbine building. Consequently, no analysis of failure modes was performed.

Due to the location of the postulated main steamline break and the location and design of the main steam and feedwater guide structures, jets from a break in the main steamline do not impact on any equipment required to mitigate the consequences of the main steamline break.

D3.6.2.1.5 Inside the Turbine Building

After the no-break zone, the piping is not seismically qualified, consequently breaks have been postulated using the fitting criteria according to the requirements of Subsection 3.6.2. Because there is no essential equipment in the turbine building and because of the installation of the third main steam isolation valve, there are no areas of concern for dynamic effects of pipe break in the turbine building.

D3.6.2.2 Feedwater System Piping

The location of the postulated pipe breaks and the pipe whip restraints for the feedwater system is shown on Figures B3.6-1 through B3.6-3. The stress analysis used for the feedwater system is summarized in Tables B3.6-1 through B3.6-3.

The essential equipment which is required for mitigating the consequences of a feedwater line break, and which may be hit by a jet from a break in the feedwater line has been evaluated for the jet impingement effects from postulated feedwater breaks. In the event of a postulated feedwater line break, any equipment which is required for the break has sufficient redundant equipment not hit by the jet or is sufficiently separated from the break so that plant safety is not affected.

D3.6.2.2.1 General

From turbine building, each of the two 20-inch feedwater lines passes through the north wall of the auxiliary building into the auxiliary building steam tunnel. Inside the auxiliary building steam tunnel, feedwater lines pass horizontally through the tunnel, through the motor-operated isolation valves, and through the air-assisted check valves into the containment building. Once inside the containment building steam tunnel, the feedwater lines, enclosed in guard pipes, pass through the tunnel into the drywell, through a check valve and a manual maintenance valve. The line then splits into two 12-inch risers which terminate at elevation 784 feet-3-1/2 inches. At this elevation termination, each feedwater line passes through the shield wall and connects to a reactor nozzle. The only other high-energy lines connecting to the feedwater lines are the 10-inch residual heat removal lines which connect inside the auxiliary building steam tunnel to the feedwater lines between the motor-operated isolation valve and the air-assisted check valves and the RHR branch lines for the FWLC mode of RHR. Refer to D3.6.2.7.6 for further description of FWLC piping.

No breaks were postulated in the feedwater system piping extending from the check valve in the drywell through the containment steam tunnel, through the auxiliary building steam tunnel up to the turbine building. Breaks were postulated only inside the turbine building steam tunnel after the first elbow fitting and inboard of the check valve in the drywell. As previously mentioned, the only other high-energy lines analyzed as part of the feedwater system are the residual heat removal 10-inch lines which connect to the feedwater lines in the auxiliary building steam tunnel between the motor-operated isolation valves and the air-assisted check valves. From the results of the pipe rupture analysis, these two 10-inch RHR lines are postulated to have no break from the feedwater line up to the auxiliary building steam tunnel wall, except that terminal-end breaks are postulated at the auxiliary building steam tunnel wall.

D3.6.2.2.2 Inside the Drywell

Pipe breaks were postulated at points of high stress or usage factor and at each feedwater connection to the reactor nozzles. To ensure that no unacceptable damage could result, restraints were installed to prevent pipe movement from damaging nearby safety-related systems, specifically the ADS system and the 10 inch RCIC steam piping which branches off the main steamline. Spray from a break inside the shield wall at the reactor pressure vessel connections is localized in the vicinity of the nozzles and poses no safety hazard. The pressurization of the annulus between the RPV and its shield wall is discussed in Subsection 6.2.1.2.3.2.

For the location of these breaks, see B3.6 Figures. Components which are in close proximity to the feedwater system risers are the ADS valves and their discharge lines, the 12-inch RHR (LPCI) injection lines Division 2, and the LPCS and HPCS injection lines. The breaks, which were postulated in the horizontal runs of the feedwater piping (outboard of the check valve and before the risers), could endanger the following safety-related lines: the 8-inch RCIC steamline, the 12-inch RHR (LPCI) injection line Division 1, and the combustible gas control system discharge lines. To preclude any likelihood of loss of a system required for safe plant shutdown, restraints and guides have been installed inside the drywell as shown on the isometric drawings.

The break of a feedwater line inside the drywell will create conditions no worse than those following a LOCA. All Class 1E electrical equipment inside the drywell whose operation, during or after a LOCA, is required for safe shutdown is qualified for the post-LOCA drywell environment as discussed in Section 3.11.

D3.6.2.2.3 Containment Steam Tunnel

All feedwater piping inside the containment steam tunnel is within the boundaries of the no-break zone and enclosed in the previously mentioned guardpipes. Consequently, a piping failure is not postulated to occur in any of these lines. Guides and restraints are located to minimize the effects of pipe movement of the feedwater pipe within the guardpipes if a break occurs outside the no-break area. The guides and restraints limit any movement to an acceptable level such that damage does not occur to either the guardpipe or any other piping in the containment steam tunnel.

D3.6.2.2.4 Outside Containment

For the feedwater piping, no breaks are postulated from the containment wall through the auxiliary building steam tunnel and into the turbine building. As previously discussed, the piping in the turbine building is not seismically analyzed and breaks are thus postulated at each fitting. There is, however, no essential equipment in the turbine building. The entire run of feedwater piping in the auxiliary building steam tunnel has no postulated breaks, therefore, no analysis of failure modes was performed. Terminal end breaks were postulated for the 10-inch residual heat removal lines analyzed as part of the feedwater system. These postulated breaks are inside the auxiliary building steam tunnel, at the wall, adjacent to each of the residual heat removal heat exchanger rooms. The effects of jet impingement and pipe whip from these breaks has been evaluated. Due to the remote location of these breaks, jets and pipe whips would not impact sufficient redundant equipment to prevent a safe shutdown of the reactor. Pressurization of the auxiliary building steam tunnel is described in Section 3.6.1, and is based on a bounding simultaneous rupture of a main steam line and a feedwater line. The reactor water cleanup lines are discussed in Section D3.6.2.8 of this attachment.

D3.6.2.3 Main Steam Isolation Valve Leakage Control System (MSIV-LCS)

Note: As a result of the re-analysis of the Loss of Coolant Accident (LOCA) using Alternative Source Term (AST) Methodology, it is no longer necessary to credit the Main Steam Isolation Valve Leakage Control System (MSIVLCS) for post-LOCA activity leakage mitigation. The system has been left in place as a passive system and is not required to perform any safety function.The composite drawings which show the piping for the main steam isolation valve leakage control system are Figures D3.6-8 and D3.6-14. The piping shown on Figure D3.6-8 is not high energy, as the isolation valves terminate the high-energy portion within the auxiliary building steam tunnel.

D3.6.2.3.1 General

The high-energy portion of the MSIV-LCS, shown on Figure D3.6-14, is that between the main steam isolation valve drain line and the isolation valves for the MSIV-LCS located just above the auxiliary building steam tunnel floor.

For the inboard MSIV-LCS, these high-energy lines consist of four 1 1/2-inch lines. For the outboard system, these high-energy lines consist of four 2-inch lines terminating in a 2 1/2-inch header which is isolated inside the main steam tunnel by normally closed motor-operated isolation valves. The size of the high-energy lines in the MSIV-LCS precludes the likelihood of their damaging any other safety-related systems in the near vicinity. Jets from postulated breaks in MSIV-LCS lines do not load any equipment required to mitigate the consequences of a break in the MSIV-LCS or affect the ability to achieve safe shutdown.

The environmental conditions associated with the breaks are the same as the local environment in the auxiliary building steam tunnel. All Class 1E electrical equipment in the steam tunnel has been qualified (refer to Section 3.11 for environmental qualification).

D3.6.2.4 Reactor Recirculation System

Reactor recirculation system Figures B3.6-29 through B3.6-31, show the locations of the postulated pipe breaks and pipe whip restraints. The stress analysis used for the reactor recirculation system is summarized in Tables B3.6-29 through B3.6-31. The piping in this system was analyzed for pipe break and pipe restraint locations by General Electric Company.

The effects of jet impingement from breaks in the reactor recirculation piping have been evaluated. In the event of a postulated reactor recirculation line break, any equipment hit by the break and required for safe shutdown has sufficient redundant equipment not hit by the jet, or is sufficiently separated from the break so that the equipment can withstand the jet forces.

D3.6.2.4.1 General

Each of the two reactor recirculation loops leaves the reactor pressure vessel at elevation 757 feet 10-1/2 inches at azimuthal angles 0 and 180 for Loops B and A, respectively. Each 20-inch suction line then drops vertically to elevation 726 the inlet isolation valve, the reactor recirculation pump, the flow control valve and the outlet isolation valve. At this point, the discharge line runs vertically up to elevation 744 feet 7-1/2 inches, where it joins the C-shaped 16-inch horizontal header. From this header, five 10-inch vertical lines run up to elevation 758 feet 3 1/2 inches, where they turn and horizontally enter the reactor pressure vessel.

In addition, from Loop B only, an 18-inch RHR suction line branches off from the vertical run between the reactor outlet and the inlet isolation valve at elevation 733 feet 6 inches. This line is considered high energy up to the isolation valve; however, it is not a part of the ECCS systems. The line passes horizontally from the connection approximately 4 feet, then turns vertically upward and rises through an isolation valve to elevation 757 feet 6 inches, where it turns and runs horizontally out of the north wall of the drywell through the containment building steam tunnel enclosed in a guardpipe and into the auxiliary building steam tunnel. This line is further discussed in Subsection D3.6.2.7.

D3.6.2.4.2 Recirculation Loop "A"

The GE analysis resulted in the postulated circumferential and longitudinal breaks shown in the Figure B3.6-31. Restraints were installed on each riser as shown on Figure B3.6-31 to limit the travel of potential ruptured piping in the direction of the RPV radius. Restraints were also installed on the 16 inch header to limit travel. In addition, one restraint was installed near the RPV nozzle of the suction line.

D3.6.2.4.3 Recirculation Loop "B"

The postulated break locations and resulting restraints are the same for both LOOPS A and B, with the exception that Loop B has the RHR connection (see Subsection D3.6.2.7.3). This connection results in the postulation of another longitudinal failure at the tee joint. To protect against this postulated failure, two additional restraints are installed, one just above and one just below the tee.

D3.6.2.4.4 Inside the Drywell

All the piping associated with the reactor recirculation system is contained within the drywell except for the RHR suction line tapping off Loop B. The risers associated with the reactor recirculation piping could conceivably damage the RHR injection lines (LPCI) and the RCIC steamline (from the main steamline to the RCIC turbine). The RHR (LPCI) lines are protected from pipe whip by restraints and guides and are redundant in function. The RCIC steamline could be impacted by a rupture of the reactor recirculation piping. However, it is protected against the dynamic effects of pipe break by restraints and guides. In addition, it is redundant in function to the high-pressure core spray system. As noted above, restraints have been installed to preclude any likelihood of the loss of a system required for safe plant shutdown.

The environmental conditions are the same as the local environment in the drywell. All Class 1E electrical equipment in the drywell has been qualified (refer to Section 3.11 for environmental qualification).

D3.6.2.5 Low-Pressure Core Spray (LPCS)

The low-pressure core spray system, Figure B3.6-5, show the locations of the postulated pipe breaks and of the pipe whip restraints. The stress analysis used for the low-pressure core spray system is summarized in Table B3.6-5.

The effects of jet impingement from breaks in the low pressure core spray piping have been evaluated. In the event of a postulated low pressure core spray line break, any equipment hit by the break and required for safe shutdown has sufficient redundant equipment not hit by the jet, or is sufficiently separated from the break so that the equipment can withstand the jet force.

D3.6.2.5.1 General

The portion of the LPCS system which is considered to be high energy is that piping between the reactor nozzle and the inboard isolation check valve. This system does not operate during normal plant operation; consequently, only that part of the piping which is normally exposed to reactor pressure is classified as high-energy.

The high-energy portion of the LPCS piping begins at the reactor nozzle at elevation 782 feet 9 inches at azimuthal angle 90°. The line passes through the shield wall penetration and drops to elevation 769 feet 5 1/4 inches, where it runs horizontally, turning at a 90° angle and passing through the locked-open maintenance valve. The line then passes through the inboard isolation valve, which is a check valve, and passes out of the drywell. No breaks are postulated after the isolation valve.

D3.6.2.5.2 Inside the Drywell

Systems which are in the vicinity of the LPCS and could be impacted by the dynamic effects of a pipe break or crack are portions of the ADS system including electrical conduits, operators and accumulators, the ADS system discharge line and the RCIC steamline. To ensure that no movement of the vertical leg can impact these systems, the vertical leg has been restrained in the event of a pipe break.

The environmental conditions are the same as the local environment in the drywell. All Class 1E electrical equipment in the drywell has been qualified (refer to Section 3.11 for environmental qualification).

D3.6.2.6 High-Pressure Core Spray (HPCS)

HPCS Figure B3.6-4 shows the locations of the postulated pipe breaks and pipe whip restraints. The stress analysis used for the HPCS system is summarized in Table B3.6-4.

D3.6.2.6.1 General

The portion of the HPCS system which is considered to be high energy is that piping between the reactor nozzle and the inboard isolation check valve. This system does not operate during normal plant operation; consequently, only that part of the piping which is normally exposed to reactor pressure is classified as high-energy.

The high-energy portion of the HPCS system begins at the reactor nozzle at elevation 767 feet 5 1/4 inches and azimuthal angle 270°. The line passes through the shield wall penetration, turns vertically downward and runs to elevation 771 feet 11 3/8 inches, where it passes through one pipe bend. From here it runs horizontally through a 90° elbow at elevation 769 feet 5 1/4 inches, turns 90° again passing through the locked open manual maintenance valve and through the inboard isolation check valve. It then makes two 75° turns and passes horizontally out of the drywell at elevation 769 feet 5 1/4 inches.

D3.6.2.6.2 Inside the Drywell

There is no essential equipment in the immediate vicinity of the HPCS piping reactor nozzle or vertical riser until the line reaches elevation 769 feet 5 1/4 inches. A break in the HPCS system piping at this elevation could impact the following systems: the ADS system discharge line, the combustible gas control system compressor discharge line and the inboard isolation valves for the drywell cooling system. To preclude the likelihood of loss of any of these systems, the piping was restrained as shown on the isometric and composite drawings.

The environmental conditions are the same as for the local environment in the drywell. All Class 1E electrical equipment in the drywell has been qualified (refer to Section 3.11 for environmental qualification).

The effects of jet impingement from breaks in the high pressure core spray piping have been evaluated. In the event of a postulated high pressure core spray line break, any equipment hit by the break and required for safe shutdown has sufficient redundant equipment not hit by the jet, or is sufficiently separated from the break so that the equipment can withstand the jet force.

D3.6.2.7 Residual Heat Removal System (RHR)

The RHR system Figures B3.6-14 through B3.6-19, B3.6-21, and B3.6-35 show the locations of the postulated pipe breaks and of the pipe whip restraints. The stress analysis used for the RHR system is summarized in Tables B3.6-14 through B3.6-19, B3.6-21, and B3.6-37.

D3.6.2.7.1 General

The piping for the RHR system is divided into three (3) parts as follows:

a. the LPCI injection lines (RHR A Loop, RHR B Loop and RHR C Loop),

b. the shutdown cooling suction line from reactor recirculation Loop B.

c. the feedwater leakage control mode (FWLC) lines to main feedwater.

Each of the above is discussed in the following subsections.

D3.6.2.7.2 Low-Pressure Coolant Injection (LPCI)

The LPCI subsystems are not in use during normal plant operation; consequently, only the piping between the reactor pressure vessel and the first normally closed valve is pressurized and thus classified as high-energy for pipe rupture analysis.

D3.6.2.7.2.1 LPCI "A"

The high-energy portion of the LPCI A Loop piping begins at the reactor nozzle at elevation 778 feet 3 1/2 inches at azimuthal angle 45°. The line passes through the shield wall penetration and drops to elevation 761 feet 3 1/2 inches. At this elevation the piping turns and passes horizontally through the locked open manual maintenance valve, the inboard isolation check valve and exits the drywell. The only safety-related systems which are in the vicinity of this loop and could be impacted by the dynamic effects of a pipe break are portions of the ADS system including the operators and accumulators and the ADS system discharge lines. To ensure that no movement of the vertical or horizontal legs can occur, restraints have been installed as shown in Figure B3.6-14.

The effects of jet impingement from breaks in the LPCI-"A" piping have been evaluated. In the event of a postulated LPCI-"A" line break, any equipment hit by the break and required for safe shutdown has sufficient redundant equipment not hit by the jet, or is sufficiently separated from the break so that the equipment can withstand the jet force.

D3.6.2.7.2.2 LPCI "B"

The high-energy portion of the LPCI B Loop, Figure B3.6-15, begins at the reactor nozzle at elevation 778 feet 3-1/4 inches at azimuthal angle 225. The line passes through the shield wall penetration and down through the locked-open manual maintenance valve and the inboard isolation check valve. Following the check valve, the line exits the drywell. No breaks are postulated after the isolation valve. There are no safety related systems in the immediate vicinity of the LPCI B Loop that could be impacted by the dynamic effects of pipe break in the high energy portion of the piping. The ADS discharge lines and the isolation valves and associated electrical conduits for the drywell cooling system will not be impacted by a break in the LPCI B Loop. Although these systems are located in this quadrant of the containment and are in the near vicinity of the LPCI B Loop piping, they are not in the vicinity of the high-energy portion. The restraints for breaks located on the high-energy portion of the piping will prevent any damage to these systems in the event of a break.

The effects on jet impingement from breaks in the LPCI-"B" piping have been evaluated. In the event of a postulated LPCI-"B" line break, any equipment hit by the break and required for safe shutdown has sufficient redundant equipment not hit by the jet, or is sufficiently separated from the break so that the equipment can withstand the jet force.

D3.6.2.7.2.3 LPCI "C"

The high-energy portion of the LPCI C Loop Figure B3.6-16 begins at the reactor nozzle at elevation 778 feet 3-1/4 inches at azimuthal angle 135°. The line passes through the shield wall and drops through the manual maintenance valve to elevation 769 feet 5 inches and through the inboard isolation check valve. Following the isolation valve, it drops down to elevation 764 feet 1/2 inch, where it passes out of the drywell. No breaks are postulated after the isolation valve. Systems which are in the vicinity of the LPCI C Loop and could be impacted by the dynamic effects of a pipe break are portions of the ADS system discharge piping, the LPCS injection line and the drywell purge system isolation valve. To ensure that no movement of the high-pressure portion of the LPCI C Loop piping can impact these systems, restraints have been installed to prevent pipe whip.

The effects of jet impingement from breaks in the LPCI-"C" piping have been evaluated. In the event of a postulated LPCI-"C" line break, any equipment hit by the break and required for safe shutdown has sufficient redundant equipment not hit by the jet, or is sufficiently separated from the break so that the equipment can withstand the jet force.

D3.6.2.7.3 RHR Suction from the Reactor Recirculation Loop B

For the shutdown cooling mode of the RHR system, Figure B3.6-17, suction is taken from reactor recirculation Loop B. This suction line begins at the tee at elevation 733 feet 6 inches and azimuthal angle 0 of the reactor recirculation loop.

The line runs vertically upward through the manual maintenance valve and through a motor-operated isolation valve. After the isolation valve, the line continues upward until at Elevation 757 feet 6 inches it turns 90° and passes out of the drywell, through a guardpipe and out of the containment. The high energy portion of this suction line is a short L-shaped run extending about 4-1/2 feet horizontally and about 14-1/2 feet vertically. Breaks were postulated as shown in Figure B3.6-17. The only system which is in the near vicinity of this suction line is the RCIC steamline. To ensure that no movement of the vertical leg of the suction line could impact this system, restraints were located to prevent pipe movement.

D3.6.2.7.4 RCIC Steamlines to RHR Heat Exchangers

Out of Service

D3.6.2.7.5 RHR Heat Exchanger Rooms

RHR Heat Exchanger Rooms A and B have been designed to withstand jet impingement and pipe whip effects of postulated breaks in the high energy RHR piping in these rooms. However, postulation of such line breaks in the RHR Heat Exchanger Rooms A and B is not required (refer to NRC letter dated January 7, 1986, Docket No. 50-461).

D3.6.2.7.6 RHR Feedwater Leakage Control Mode (FWLC) Piping to Main Feedwater

The FWLC piping of the RHR system is not in use during normal operation. The portions of piping from the RHR pump rooms to the check valves in the auxiliary building steam tunnel are classified as moderate - energy piping per Section 3.6.2.1.4.d. Consequently, only the piping between the main feedwater headers and the first check valve in each line is classified as high enery piping. The two portions of piping (3/4 inch) connected to the feedwater header down stream of valves 1B21-F032A/B are considered to be within the break exclusion boundary for piping between containment isolation valves. Therefore no pipe breaks are postulated in these sections of piping. The two portions of piping (2 inch) connected to the feedwater header up stream of valves 1B21-F032A/B are postulated to have terminal end breaks at the connection to the main feedwater header. The pipe whip and jet impingement resulting from these breaks have been evaluated and do not affect any adjacent components or equipment in the steam tunnel.

D3.6.2.8 Reactor Water Cleanup System (RWCU)

The reactor water cleanup system isometric drawings, Figures B3.6-23 through B3.6-30, show the location of the whip restraints and associated postulated pipe breaks. The stress analysis used for the reactor water cleanup system is summarized in Tables B3.6-23 through B3.6-30.

D3.6.2.8.1 General

For many purposes, the reactor water cleanup system can be considered as two nearly separated subsystems: (1) the loop from the reactor, through the heat exchangers, through the filter demineralizers, back through the regenerative heat exchangers, and return to the reactor through the feedwater line; and (2) the auxiliary subsystem which removes the used demineralizer resin and replaces it with new resin. The first subsystem is almost entirely classified as a high-energy system; the second subsystem, even during the small time it is in operation, is classified as moderate energy. Neither is required for a safe plant shutdown. From the standpoint of piping failure, the only concern is the possible detrimental effect of a RWCU pipe rupture on other equipment required for safe shutdown.

One 4-inch line taps off of the bottom of each reactor recirculation loop at azimuthal angles 155° and 335°. Each line runs axially out from the reactor at elevation 724 feet 8 inches. Each line passes through a motor-operated isolation valve, turns and runs vertically upward to elevation 732 feet. At this point, both lines turn and circle towards one another around the inside of the weir wall, meeting at a tee at an azimuthal angle of approximately 58°. Two feet before the tee, both lines increase in diameter to 6 inches and, from the tee, the combined suction 6-inch line turns and runs back to azimuthal angle 19.6.

At this point, the pump suction line drops horizontally to elevation 725 feet 4 inches, turns, and runs through a motor-operated isolation valve. Immediately after the motor-operated isolation valve, the line tees with the pump suction line from the bottom of the reactor vessel. The pump suction line from the bottom of the reactor vessel is routed from elevation 742 feet 6 inches at zimuthal angle 210 northward to where it drops to 725 feet 4 inches and joins with the pump suction line coming from the reactor recirculation lines. For the actual routing of this reactor drain line, see Figure B3.6-23.

One line branches off of the pump suction line from the bottom of the reactor vessel. It contains a normally closed motor-operated bypass line used in the hot standby mode. Another branch from the bypass is a 2-inch reactor drain line which runs to a sump in the drywell. The combined pump suction line (consisting of the combined line from the reactor recirculation loops and the line from the reactor pressure vessel) runs vertically upward from elevation 725 feet 4 inches to elevation 756 feet 5 inches. There the line turns and runs horizontally through the inboard motor-operated containment isolation valve. It then exits the drywell, passing through the containment inside the containment steam tunnel enclosed in a guardpipe.

Upon exiting the containment, the line passes through the outboard containment motor-operated isolation valve, turnsand is routed around the outside of the containment wall to the reactor water clean up pump room cubicles. These are located on elevation 737 feet in the area bounded by column rows 117-123 and column rows Z-AB. The pump discharge line then returns to the auxiliary building steam tunnel (along the same general routing), where it passes through the outboard motor-operated containment isolation valve into the containment building steam tunnel. Then the line passes through the inboard containment isolation valve and runs up to the floor above the steam tunnel where the reactor water cleanup heat exchangers are located.

Discharge lines from heat exchangers join together in one line which goes through the steam tunnel and then splits into two lines each terminating in separate RWCU filter demineralizers. They are located in cubicles at azimuth 270 and elevation 803 feet 3 inches.

Discharge lines from filter demineralizers go back to the steam tunnel in the containment building and both are connected to one pipe. This pipe passes through the inboard containment isolation valve and out of the containment by way of the outboard containment isolation valve. At this point, the line is no longer considered high-energy; however, it does continue northward out of the auxiliary building steam tunnel, into the turbine building and goes to the main condenser. The elevation at which this penetration exits the containment is 762 feet 3 inches.

The second line returns reactor water from the reactor water cleanup system to the feedwater system. Headers from the heat exchanger drop vertically downward into the containment building steam tunnel, along the same path as the reactor water cleanup line to the condenser. At elevation 763 feet 8 7/8 inches, the line passes through the inboard containment isolation valve and through the outboard containment isolation valve, where it tees into two branch lines, each routed to one of the feedwater lines. One leg of the tee runs westward through a motor-operated isolation valve and becomes the RHR line (at elevation 763 feet 8-7/8 inches) which terminates in the feedwater line. The other branch of the tee runs across the steam tunnel, where it turns southward through a motor-operated valve and becomes the RHR line which terminates in the second feedwater line.

D3.6.2.8.2 Inside the Drywell

In the pump suction lines, circumferential breaks have been postulated as shown in Attachment B3.6. The dynamic load of a ruptured RWCU line could impact several systems:

the RHR system (suction line for shutdown cooling mode), the main steamlines, and the feedwater lines.

The RHR line itself, an 18-inch line, will not be affected by the rupture of a 6-inch RWCU line. Also in this same area are the main steam and feedwater lines. Both are of such size that they would not be affected by the rupture of the 6-inch RWCU line.

The environmental conditions in the drywell are the same as the local environment. All Class 1E electrical equipment in the drywell has been qualified (refer to Section 3.11 for environmental qualification).

The effects of jet impingement and pipe whip from postulated breaks in the RWCU lines have been evaluated. Due to the small size and remote location of the RWCU lines, jets and pipe whip from any RWCU line break do not hit sufficient redundant equipment to prevent safe shutdown of the reactor.

D3.6.2.8.3 Inside and Outside Containment

The RWCU piping is located in the following areas:

1. Auxiliary Building Main Steam Tunnel;

2. Auxiliary Building RWCU Pump Cubicles;

3. Auxiliary Building Radwaste Pipe Tunnel;

4. Containment RWCU Cubicles and Tunnels;

5. Containment Main Steam Tunnel.

The effects of jet impingement and pipe whip from postulated breaks in the RWCU lines have been evaluated. Due to the small size and remote location of the RWCU lines, jets and pipe whip from any RWCU line break do not hit sufficient redundant equipment to prevent safe shutdown of the reactor.

D3.6.2.9 Standby Liquid Control System

The postulated pipe breaks for the standby liquid control system are shown in Figure B3.6-33.

D3.6.2.9.1 General

The high-energy portion of the standby liquid control system is that which is shown on Figure B3.6-33. The high-energy portion is between the connection to the reactor pressure vessel at elevation 742 feet 3 inches and the first check valve. The line runs from the bottom of the reactor pressure vessel at azimuthal angle 225° outward axially approximately 1 foot, where it drops to elevation 743 feet 3 inches, turns and passes out through the shield wall at elevation 738 feet. Once outside the shield wall, the piping turns and runs vertically upward to elevation 741 feet, where it passes through a motor-operated isolation valve and a check valve. This check valve will terminate the high-energy portion of the line analyzed for pipe rupture.

D3.6.2.9.2 Inside the Drywell

The only systems within the immediate vicinity of the standby liquid control system high-energy piping are the HPCS system, the reactor recirculation system and the control rod drive system. The 20-inch reactor recirculation line and the 10-inch HP line will be unaffected by the rupture of the 3-inch standby liquid control system line. The standby liquid control line is routed such that a break within the shield wall cannot impact the control rod drive system insert or withdrawal lines.

The environmental conditions are the same as the local environment in the drywell. All Class 1E electrical equipment in the drywell has been qualified (refer to Section 3.11 for environmental qualification).

The effects of jet impingement and pipe whip from potential breaks in the SLCS piping have been evaluated. Due to its routing and small size, jets and pipe whip from postulated breaks in the SLCS do not hit sufficient redundant equipment to prevent safe shutdown of the reactor.

D3.6.2.10 Reactor Core Isolation Cooling System (RCIC)

The RCIC isometric drawing shown in Figures B3.6-20 through B3.6-22 show the postulated pipe breaks in this system. The stress analysis used for the reactor core isolation cooling system is summarized in Tables B3.6-20 through B3.6-21 and for subsystem RI-11, the stress analysis is summarized in calculation EMB-049168.

D3.6.2.10.1 General

Two portions of the RCIC system piping are considered to be high-energy: (1) the steamline to the RCIC turbine, and (2) the reactor pressure vessel head spray line.

An 8-inch branch line from main steamline A delivers steam to drive the RCIC turbine. The 8-inch line branches off main steamline A at elevation 788 feet 3 3/4 inches. The line circles around the outside of the shield wall to azimuthal angle 0°, where it drops vertically downward to elevation 761 feet 5 inches. After two turns, the line passes through the normally open inboard containment isolation valve and drops to elevation 758 feet 4 3/8 inches, where it passes horizontally out of the drywell and through the containment building steam tunnel. The line exits the containment building steam tunnel and passes through the outboard motor-operated containment isolation valve in the auxiliary building steam tunnel. After the outboard containment isolation valve, the line runs northward approximately 19 feet. The original line reduces in size from 8 inches to 4 inches. The 4-inch line turns vertically downward and drops from elevation 757 feet 6 1/2 inches through the auxiliary building steam tunnel floor to elevation 730 feet, where it goes into the RCIC turbine cubicle. Jets and pipe whip from postulated breaks in the RCIC turbine steam supply line do not impact any equipment required to mitigate the consequences of the break. The compartment at elevation 707 feet 6 inches is the RCIC turbine cubicle. All equipment in this area is for support of the RCIC turbine.

The portion of the RCIC system connected to the reactor pressure vessel head spray which is considered high-energy is that portion between the reactor pressure vessel nozzle and the first check valve. The vessel nozzle is located at elevation 813 feet 10 3/8 inches. Welded to the nozzle is a 6-inch tee. One branch of the tee (at azimuthal angle 150°) is connected to a 4-inch pipe that is reduced to 2-inch size and connected to main steamline "A" at elevation 790 feet 6-1/2 inches. The function of this line is to vent noncondensable gas from the reactor pressure vessel head. The other branch connection located at azimuthal angle 330°, is the inlet to the

RPV head spray line from the RCIC pump. After the tee, this line reduces from 6 inches to 4 inches and passes through a check valve. This terminates the portion of the line considered high-energy for pipe rupture.

D3.6.2.10.2 Inside the Drywell

In the immediate vicinity of the steamline to the RCIC turbine are the 3-inch main steamline drains, the 12-inch LPCI A injection line, and the 12-inch feedwater line. The dynamic load of the steamline to the RCIC turbine could not affect the larger LPCI injection line or the feedwater line. However, it could impact the 3-inch main steamline drains. The main steamline drains are not required to mitigate the consequences of an accident. To limit the movement of the vertical run and thereby the moments on the inboard containment isolation valve, restraints were located on both the vertical run and the horizontal run inboard of the inboard containment isolation valve. Outboard of the containment isolation valve between the valve and the guardpipe, restraints were located to prevent unacceptable movement of that piping and conceivably impairment of operation of the isolation valve.

Also, in the drywell is the RCIC discharge to the reactor pressure vessel head spray. This line is fitting-to-fitting from the reactor nozzle to the inlet of the check valve. The actual routing is nozzle to tee to reducer to back-to-back 45 degree bends to check valve. The RCIC system Figure B3.6-22 shows the location of the postulated pipe breaks and the pipe whip restraints. The actual length of piping precludes damaging any safety-related component in this area. Of primary concern in this area is the drywell head. The drywell head is designed to accommodate jet impingement and pipe whip loads from the RCIC head spray line.

The environmental conditions from either a reactor pressure vessel head spray line break or a break of a steamline to the RCIC turbine line are the same as the local environment in the drywell. All Class 1E electrical equipment in the drywell has been qualified (refer to Section 3.11 for environmental qualification).

The effects of jet impingement and pipe whip from potential breaks in the RCIC piping have been evaluated. Both the RCIC steam supply and head spray lines are routed such that jets from potential breaks in these lines do not hit sufficient redundant equipment to prevent safe shutdown of the reactor.

D3.6.2.11 Control Rod Drive System

Breaks are not postulated in the CRD lines. As noted in Subsection 3.6.2.1.4f, CRD insert lines are exempted. Breaks are not postulated in the CRD withdrawal lines due to the small size and energy content of the lines.

D3.6.3 MODERATE ENERGY PIPING

Through-wall leakage cracks were postulated to occur in accordance with Subsection 3.6.2. These cracks were assumed to result in wetting of any equipment in the area whether above or below the crack and at substantial distances from the crack. In addition, the possibility of compartment flooding was considered and is discussed in Subsection D3.6.4.

Each room, compartment and/or area in each seismic Category I building has been evaluated. Class 1E electrical components are not evaluated for the spray effects associated with a postulated moderate energy line crack in areas where the environmental conditions of a postulated high energy line break exist. The environmental conditions of a high energy line break are more severe than for a moderate energy line crack. Class 1E electrical components are qualified for the environmental conditions (steam, 100% relative humidity and condensation) resulting from a postulated high energy line break. This environmental condition (with associated pressure) envelopes the spray effects of a moderate energy line crack.

In areas where the environmental conditions caused by a high energy line break do not exist, the Class 1E electrical components were conservatively assumed to become nonfunctional due to a moderate energy line crack (spray) with the exception of valve actuators, junction boxes, pullboxes, conduits, cable trays, limit switches, solenoid valves and instruments which are environmentally qualified for steam, 100% relative humidity and condensation.

Each area of the auxiliary building, control building, diesel generator building, fuel building, and circulating water screenhouse were evaluated for moderate energy line cracks. These areas can be identified by referring to the general arrangement drawings cited in Chapter 1 and the composite drawings in this attachment. The drywell and certain areas of the containment are subject to high energy line breaks and were not evaluated for moderate energy line cracks.

Concurrent with the moderate energy line crack, a single failure and SSE were postulated. For some systems (e.g., RHR and shutdown service water), a single failure in the redundant system has been excluded in accordance with Subsection B.3.b(3) of BTP ASB 3-1. In all such cases the redundant system train meets the qualification of that NRC position. Loss of offsite power was assumed where a turbine generator or reactor protection system trip is a result of the postulated line crack. Using the above postulates and assumptions, an analysis was done which identified all equipment in the same room as a MELB and which may be required for plant shutdown. A single failure analysis was performed to determine if the identified equipment was required for safe shutdown or to mitigate the consequence of the cracks in question. The results of this analysis are summarized below.

D3.6.3.1 Containment

All equipment within the containment and drywell which must operate during or after a LOCA is qualified for the appropriate accident environmental conditions as described in Section 3.11. The wetting associated with a postulated failure of any moderate energy piping is within the bounds of that qualification.

D3.6.3.2 Auxiliary Building

Each compartment or significant area is discussed in detail starting at the lowest elevation, 707 feet 6 inches. Most of the essential equipment is located on the lower two levels in compartments north of the containment.

D3.6.3.2.1 RHR Pump Room A

RHR pump room A, including the compartment housing RHR heat exchanger A, is located on elevation 707 feet 6 inches between column rows 114, 121 and U,AA. It has been assumed that any piping failure in this room would disable all the equipment in the room. This would result in the loss of the RHR pump A and its associated support equipment, such as the instrument panel and area cooling coil cabinet. Loss of this equipment would not impair in any way the RHR B system, the RHR C system, or the reactor core isolation cooling system. All other ECCS would remain operable, subject to the additionally postulated single active failure. The consequences would remain within the range of events analyzed in Chapters 6 and 15.

The electrical equipment is not adversely affected due to wetting caused by a MELB spray when the equipment in the room is qualified for 100% humidity, which is no worse than the wetting of the equipment due to a MELB.

Also of concern in this area are several containment isolation valve assemblies. These valve assemblies are qualified to withstand a water spray and still remain functional. However, even should any or all of these valves fail in the worst position, no significant release of radioactivity would occur. The standby gas treatment system and the auxiliary building gas control boundary would still remain functional.

D3.6.3.2.2 RHR Pump Room B

The RHR pump room B is located on elevation 707 feet 6 inches between column rows 105, 110, and U,AA. Because of the similarity between the arrangements of RHR A and RHR B pump rooms, the basic analysis and conclusions of RHR A (listed in the preceding subsection) apply.

D3.6.3.2.3 RHR Pump Room C

RHR pump room C is located on elevation 707 feet 6 inches between column rows 102, 105 and U.8,AB. All of the equipment in this room is associated with the RHR C pump, except for some leak detection and containment monitoring instrumentation. Therefore, because of the similarity of RHR pump room C and RHR pump rooms A and B, the basic analysis and conclusions drawn previously apply.

D3.6.3.2.4 RCIC Pump Room

The RCIC pump room is located on elevation 707 feet 6 inches between column rows 110, 114 and U,Z. It was assumed that all equipment in this room would be inoperable from the effects of pipe rupture spray. The only equipment in this room is that associated with the RCIC system, which is not required for safe shutdown or the mitigation of the effects of any piping failure in this room. Failure of the RCIC system would not prevent safe shutdown of the reactor.

D3.6.3.2.5 LPCS Pump Room

The LPCS pump room is located on elevation 707 feet 6 inches between column rows 121, 124 and U.8, AA. All of the equipment in this room is associated with the LPCS, except for some leak detection, containment monitoring and suppression pool monitoring instrumentation. This instrumentation, as well as all electrical circuits and all piping in this area, are primarily Division 1 except for a few Division 2 components that are required. The electrical equipment is not adversely affected due to wetting caused by a MELB spray when the equipment in the room is qualified for 100% relative humidity, which is no worse than a wetting of the equipment due to a MELB. A failure of LPCS or a failure of Division 1 is compensated for by the use of Division 2 RHR LPCI Loops B and C.

D3.6.3.2.6 MSIV Leakage Control System Rooms

The MSIV leakage control system rooms are located on elevation 737 feet 0 inch between column rows 110, 112, and W,Z. The inboard system (Division 1) and the outboard system (Division 2) are physically separated from one another by a barrier wall halfway between column rows 110 and 112. This wall physically ensures that any steam or water line break in one room will affect that division only, leaving the other division to function. It was conservatively assumed that a water spray from a broken line in either room would disable all the equipment in that room.

D3.6.3.2.7 Pipe Tunnel

The auxiliary building tunnel is a vertical tunnel that passes through elevation 737 feet 0 inch from the auxiliary building steam tunnel to the RCIC pump room. The tunnel is located between column rows 110, 114 and U,U.8. This pipe tunnel contains high energy piping and the valves associated with operation of the RCIC turbine. There are no other safety-related components in this tunnel.

D3.6.3.2.8 Auxiliary Building Steam Tunnel

The auxiliary building steam tunnel was not evaluated for a MELB, since HELBs are postulated in this area. The condition of a HELB is more severe than that of a MELB.

D3.6.3.2.9 Electrical Switchgear/Motor Control Center Rooms

Two rooms in the Auxiliary Building Elevation 781'-0" contain Divisional Class 1E switchgear and motor control centers. The room to the West of the Auxiliary Building Steam tunnel is located between Column Rows 102, 107, S and AD; and contains Division 2 equipment. The room to the East of the Auxiliary Building Steam tunnel is located between Column Rows 117, 124, S and AB; and contains Division 1 equipment. The only piping in these areas that is of concern, is the shutdown service water piping supplying the room area coolers. This portion of the piping in the area of the switchgear has been classified as a no-break area due to the low piping stresses.

D3.6.3.3 Fuel Building

The fuel building contains relatively little electrical equipment to be protected from a water spray. The only essential equipment in the fuel building is in the high-pressure core spray cubicle containing its associated support equipment. The other item in this area which is supplied with electric power is the low-frequency motor generator sets, which supply electrical power to the reactor recirculation pumps. This item is not required for the safe shutdown of the plant and was therefore not evaluated for damage from water spray.

D3.6.3.3.1 High-Pressure Core Spray Pump Room

The high-pressure core spray pump room is located on elevation 707 feet 6 inches between column rows 102, 106 and AD,AH. All of the equipment in this room is associated with the high pressure core spray system which is Division 3. The electrical equipment of HPCS is not adversely affected by wetting caused by a MELB spray when the equipment in the room is qualified for 100% relative humidity, which is no worse than the wetting of the equipment due to a MELB. Even assuming a failure, the plant can be safely shut down without the use of the high-pressure core spray.

D3.6.3.4 Control Building

Piping in the control building was located with the intention of minimizing the probability of equipment damage from pipe leaks. This approach, combined with the special separation of redundant equipment in the system, results in very low probability of any hazardous effects of leakage. The plant service water and chilled water systems are the primary sources of wetting in the control building. The chilled water lines are located in the basement of the control building. The hydrogen recombiner skids are the only essential equipment located in the basement area. The recombiners are designed to remain operable within the environment of a water spray.

Piping on other floors is limited to plant service water and cooling water for area coolers and chillers. The lines are routed to the extent feasible in areas where there is no other equipment. Therefore, a postulated failure could not wet essential switchgears, motor control centers or equipment. No piping is located in or near the main control room or in the cable spreading room below the control room.

D3.6.3.5 Diesel-Generator Building

Each of the two standby diesel generators and the one HPCS diesel generator has its own room within the building. The existence of doors between the divisional diesel generator rooms has no safety significance. If a postulated line break occurs in the shutdown service water piping (which is the largest fluid source in the room), the floor drain system provided in the diesel generator cubicles is sized to accommodate the expected flow from the pipe break. The doors have an 8" curb. No water accumulation would occur such that the remaining other divisional equipment could be impaired. Consequently, a moderate energy line break in the piping for any one diesel and a single failure will leave sufficient redundant sources of power to safely shut down the reactor.

In addition, most of the piping is located relatively low in the building, and by the arrangement of the electrical equipment, sensitive items would not be significantly wetted.

D3.6.3.6 Circulating Water Screen House

The circulating water screen house contains the shutdown service water pumps of Divisions 1 and 2 and the cooling water pump for the high-pressure core spray diesel. These pumps and their associated equipment are the only essential items located in the circulating water screen house.

The shutdown service water pumps are located on elevation 699 feet. The pumps for Unit 1 are located in the northeast corner of the building between column rows 1 and 2, close to column C. Each pump is located in its own cubicle and physically separated from all other pumps. All associated support equipment for each pump is located in its respective cubicle. Consequently, no postulated pipe failure in either pump room would disable the redundant pump for that unit. In addition, failure of the Shutdown Service Water System (Division 3) to supply cooling water to the high-pressure core spray diesel cooling system would disable only the high-pressure core spray diesel. This would not prevent safely shutting down the plant.

D3.6.4 INTERNAL FLOODING

An analysis has been performed to ensure that flooding as a result of postulated high and moderate energy line breaks will not compromise the safe shutdown capability of the Clinton Power Station. Flood levels for various areas of the plant were calculated. These flood levels were used as input to structural load and safe shutdown assessments. See Subsections 3.4.1 and 3.8.4 for structural load assessment due to flooding.

Safe shutdown for postulated flooding due to internal piping failures is analyzed for areas containing safe shutdown equipment and/or areas where flooding could potentially impact associated circuits of electrically operated safe shutdown equipment. The buildings which were analyzed for safe shutdown following postulated internal flooding are:

a. auxiliary building,

b. containment (including drywell),

c. control building,

d. diesel-generator building,

e. fuel building,

f. radwaste building,

g. screenhouse, and

h. turbine building.

The auxiliary, containment, control, diesel-generator, fuel, and screenhouse buildings contain safety-related equipment. The flood protection arrangement of the Circulating Water Screen House is shown in Figure D3.6-134. The radwaste and turbine buildings do not house safety-related equipment, but flooding in these buildings has potential impact on buildings which do house safety-related equipment.

Analyses were performed to determine the flood level response of the various areas of the station to postulated failures of moderate and high energy fluid systems. To accomplish this, the station was divided into flood zones.

Many of these zones, termed "general areas," are areas that exhibit large open spaces within the plant and often contain stairwells and hatches that are open to lower levels. Other areas, termed "subcompartments," are smaller areas (generally) enclosed by Seismic Category 1 walls that open to the general areas only through doorways or hatches.

Maximum flood levels for any particular flood zone were calculated assuming a single piping failure as the initiating event (Branch Technical Position ASB3-1).

For each zone, high and moderate energy piping was evaluated to determine which single postulated line failure would produce limiting flood levels. The fluid release rates were calculated based on crack or break sizes determined from Standard Review Plan 3.6.2 and the duration of the release was generally taken to be 60 minutes. Duration of the release was generally taken to be 46 minutes for the circulating water expansion joint failure and 30 seconds for isolable RWCU failures. Certain breaks (e.g., ECCS suction lines) were assumed to be non-isolable. In most cases (i.e., MELB) the blowdown rate was considered to be constant. In other cases (i.e., HELB) an initial high blowdown rate (based on line inventory) was followed by a smaller rate based on an upstream limiting area. Potential/pipe and jet/impingement piping failures following a HELB could contribute to the flood source and these were considered in the analysis.

Fluid removal from general areas was by means of centrally located stairwells or open hatches and to a lesser extent, floor drains. Where no such removal paths were present, the fluid was assumed to accumulate within the area. Fluid removal from subcompartments was by means of floor drains and flow under doors leading to general areas.

Flood levels for the zones were calculated for breaks within and outside of the areas of interest and the limiting flood levels were tabulated. The calculated flood levels were found to be less than 2 inches for most areas of the plant. The zones that experience flood levels exceeding 10 inches are concentrated in the lower elevations of the plant while flood zones in the upper floors of the plant outside containment would not experience flooding above 10 inches. This result reflects the general design of systems within the plant with large lines and high energy systems located primarily on the lower floors.

One potential source of major flooding is a postulated failure of the circulating water expansion joint in the turbine building. In the event that the non-safety-related flood mitigation systems fail to perform their function, a postulated failure of this expansion joint could result in flooding up to elevation 719 feet in all areas of the power block except for the auxiliary building, the HPCS pump room, and the diesel oil tank rooms.

Another major source of flooding is the postulated failure of any of the non-isolable portions of the ECCS pump suction lines to the suppression pool. Postulated failure of one of these lines could result in flooding of a single ECCS cubicle below the cubicle watertight elevation (elevation, 731 feet 5 inches).

Flooding events are analyzed by one of two methods for the safe shutdown assessment:

a. zone-by-zone basis

b. flooding source-by-source basis

The zone-by-zone method examines the maximum flood level in each zone (for any postulated piping failure) and determines which, if any, electrically operated safe shutdown equipment is subject to submergence. This method is applied to all areas of the plant except for portions of the lowest elevation (elevation, 707 feet 6 inches/712 feet) in the auxiliary building.

The source-by-source method examines safe shutdown on a flood source-by-source basis. For each postulated piping failure, the effect of the calculated flood level in that flood zone as well as concomitant levels in other zones is evaluated for safe shutdown. In addition to identifying which, if any, safe shutdown equipment is subject to submergence, the potential for reactor/turbine trip as a direct consequence of piping failures was also evaluated. This source-by-source method is applied to portions of the lowest floor (elevation, 707 feet 6 inches/712 feet) of the auxiliary building.

For each flooding event, the most restrictive SAF was assumed. When the flooding event was due to a postulated failure in a dual purpose moderate energy piping system, a SAF in the redundant system was not considered per Paragraph B3.b(3) of BTP-ASB 3-1 of Standard Review Plan 3.6.1. Credit for use of the redundant dual-purpose system for achieving safe shutdown is contingent upon the availability of a diverse power supply (i.e., offsite power and emergency onsite power). This dual purpose exclusion criteria is applied to the RHR heat exchanger trains (RHR-A, RHR-B) and the directly associated supporting systems which include the shutdown service water train, the cubicle coolers, and the Class 1E divisional power supply.

Loss-of-offsite power is assumed when reactor/turbine trip is a direct consequence of the initial flooding event. In certain instances, LOOP is not a conservative assumption, and thus was not assumed (e.g., the postulated circulating water expansion joint rupture in the turbine building requires offsite power to drive the flooding source - the circulating water pumps).

Postulation of line breaks in high energy RHR piping in the RHR Heat Exchanger Rooms A and B is not required (refer to NRC letter dated January 7, 1986, Docket No. 50-461). Therefore, flooding from these sources is not included in the design basis. Flooding from postulated high energy line breaks within the Auxiliary Building Main Steam Tunnel have been included in the design basis, and safe shutdown has been demonstrated for these postulated breaks.

Safe shutdown has been assured by analysis for design basis flooding events.

The height of the stairway landing/door bottom (elevation 715 feet) between the turbine building (elevation 712 feet) and the auxiliary building (elevation 707 feet 6 inches) is above the water level in the turbine building that would be caused by the unlikely rupture of a line from the two condensate storage tanks and the consequent release of 800,000 gallons of water into the turbine building. In the event of a simultaneous rupture of this line and a line from the demineralized water storage tank (400,000 gallons), water will not be contained in the turbine building. However, water will not enter the core spray cooling pump (RHR, LPCS, HPCS, and RCIC) cubicles in the auxiliary building, since they are watertight to elevation 731 feet 5 inches.

In the event that an entire circulating water expansion joint fails, leaving a 6.25 inch gap between the piping and the waterbox, 267,400 gallons of water per minute will be released to the turbine building. Each condenser cavity, designed to contain flooding to elevation 715 feet, is equipped with a redundant system of level switches which will alarm in the control room if the water level in the condenser cavity reaches an elevation of more than 1 foot (elevation 710 feet) above the condenser cavity floor at elevation 709 feet. Additionally, these level switches will close a motor-operated valve in the floor drain piping between the condenser cavity and the turbine building floor drain sump to slow flooding of the turbine building. A second system of redundant level switches will automatically stop the circulating water pumps if the flood water reaches an elevation of 714 feet within the condenser cavity. An additional foot, from elevation 714 feet to elevation 715 feet remains to contain the water flow due to the coastdown of the circulating water pumps after they are initially signaled to stop. The level switches that stop the circulating water pumps are powered by CW Pump A control power. Therefore, if the CW Pump A control fuses are pulled, these level switches would not stop CW Pumps B & C, if required.

In the event of failure of an expansion joint and both redundant sets of level switches, the turbine building could be flooded above 715 feet. Then, because of flow areas between the turbine building and radwaste and control buildings, they could be flooded also. The limiting level of 719 feet could be reached only after 46.6 minutes with no operator action to stop the circulating water pumps. The turbine building water level could reach 726.7 feet in this time, but no essential equipment would be affected. In addition, even assuming the failure of level switches, postulated above, the control room operator will still have adequate warning of flooding in the turbine building. By elevation 719, the CRD pumps and turbine building MCC's 1A, 1B, 1C and 1D, among other things, will be flooded, and before 726.7 feet the condensate and condensate booster pumps will be lost. The control building is protected up to 719 feet. The radwaste building contains no equipment required for safe shutdown.

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. All 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 horizontal design response spectra defined at the ground surface (surface spectra) are shown on Figure 3.7-12 through 3.7-15 and Figures 3.7-20 through 3.7-23 for OBE and SSE, respectively. The vertical design response spectra defined at the ground surface (surface spectra) are shown on Figures 3.7-16 through 3.7-19 and Figures 3.7-24 through 3.7-27 for OBE and SSE, respectively. The maximum horizontal and vertical ground accelerations at the foundation level corresponding to the above site response spectra are 25% of gravity for the safe shutdown earthquake (SSE) and 10% of gravity for the operating basis earthquake (OBE). The design response spectra comply with Regulatory Guide 1.60 with clarificaiton as discussed in 2.5.2.6 and 2.5.2.7.1.

In accordance with agreements reached on September 24 and October 15, 1981 meetings between IP and the NRC staff, IP performed a seismic hazard analysis for the CPS site to demonstrate that the plant is located in an area which does not pose any higher seismic hazard than anywhere else in the Central Stable Region. Consistent with the results of this hazard analysis, site specific response spectra will be developed which will be used as the basis for comparing response spectra (derived from a soil spring analysis) to the current design response spectra which were derived from a finite element analysis. This comparison is expected to show that the finite element analysis results either bound or are very close to the soil spring analysis results.

In order to resolve the concern identified in this question, as well as in questions 220.15, 220.21, and 220.26, the following work has been done:

a. Site specific response spectra were developed for the Clinton site for a 5.8 magnitude earthquake, as described in the Weston Geophysical report, "Site Specific Response Spectra for the Clinton Power Station - Unit 1", submitted to the NRC staff with References 16 and 17.

b. The soil-structure interaction analysis was performed using the soil spring method; variation in soil properties was also considered in this analysis; and no deconvolution was used.

c. The critical plant structures, piping and equipment were reevaluated to the new seismic loads, and it was concluded that the Clinton plant design is conservative and can withstand the new seismic load.

d. The details of the analysis described in paragraph (b) above, the responses obtained from these analyses, and the reevaluation results are contained in References 18, 19 and 20. (Q&R 220.14)

3.7.1.2 Design Time History

The following two-step procedure is used for generating the foundation and rock level (base on the soil-structure interaction system) time histories. In the soil-structure interaction analysis, the rock time history is applied at the base of the soil-structure model.

STEP 1: Generation of Design Time History

The north-south and vertical components of the 1940 El Centro earthquake records are modified using the program RSG (see Appendix C for description of RSG) so that the response spectra generated using these synthetic records match closely the Regulatory Guide 1.60 response spectra for the horizontal and vertical directions.

The frequencies used in generating the response spectra from the modified synthetic time histories are spaced as follows:

Frequency Range (Hz)	Increment (Hz)
0.5-3.0	0.10
3.1-3.6	0.15
3.6-5.0	0.20
5.0-8.0	0.25
8.0-15.0	0.50
15.0-18.0	1.0
18.0-22.0	2.0
22.0-34.0	3.0

The comparison of response spectra obtained from horizontal and vertical synthetic time histories and the corresponding Regulatory Guide 1.60 design spectra for 0.2g ground acceleration is presented in Figures 3.7-1 through 3.7-10 for 2, 3, 4, 5, and 7 percent damping ratios. The synthetic time histories are then scaled to the required ground surface design response spectra acceleration levels.

STEP 2: Generation of Foundation and Rock Motion

The soil profile above the rock is modeled as a one-dimensional continuous shear layer system. The layering scheme of the 246 feet of soil below the plant complex is shown in Figure 3.7-11.

The soil below the circulating water screen house is shown in Figure 3.7-76. The design time history obtained in Step 1 is applied at the ground surface, and the foundation and rock time histories are obtained using the program SHAKE (see Appendix C for description of SHAKE program). The strain-dependent soil properties for various soil layers used in the SHAKE analysis are given in Table 3.7-9. A comparison of the free field foundation and design spectra for the main plant for 1, 2, 3, and 4 percent damping ratios for OBE and 1, 3, 4, and 7 percent damping ratios for SSE are given in Figures 3.7-12 through 3.7-27.

Variation in soil properties at the site has been taken into account in the soil-structures interaction analyses using soil spring method. A detailed review of the dynamic soil properties was undertaken. The goal of the review was to define the upper and lower bound curves of soil shear modulus values. At the same time it was decided to develop site specific response spectra for the Clinton site to resolve the entire seismic soil-structure interaction issue identified by NRC staff in questions 220.14, 220.21, and 220.26. For this purpose, an estimate of shear wave velocities for soils present below the foundation mat was required. A review of the shear wave velocities give in Figures 2.5-369 through 2.5-371 suggested that in light of the knowledge gained from comparable soil deposits, the shear wave velocities given in these figures were high. The shear wave velocities given in the FSAR were computed from the measured compressional wave velocities and estimated Poisson's ratio. In view of the current knowledge, the estimated values of Poisson's ration are considered low.

Based on the above, a thorough review of the shear wave velocities and the low-strain soil moduli was performed by Dames & Moore. Based on the results of this review, Figures 2.5-369 through 2.5-371, and Table 2.5-46 and 2.5-48 were revised in Amendment 12, dated January 1982.

The rationale and references that were used to estimate the shear velocity for the glacial soils are summarized in Table A3.7-1, Evaluation of Geophysical Data. This table summarizes the geophysical measurements made at five nuclear plant sites. In all cases, both the compressional and shear wave velocities were measured in the field. As shown in the table, Poisson's ratios for glacial soils range from 0.45 to 0.48 based on the values calculated from the measured velocities presented in the table. As a result, it was estimated that Poisson's ratios for the Illinoian glacial till and Wisconsinan glacial till are 0.46 and 0.48, respectively. These values, along with the measured compressional velocity, were utilized to calculate the estimated shear wave velocity at Clinton.

For structural fill, the normalized shear modulus factor (k2) versus shear strain relationship was established from the laboratory test data. In addition, k_{2max} was calculated using the Hardin and Drnevich equation, Reference 21. Since the k_{2max} value obtained from the lab data is less than the k_{2max} value calculated from the Hardin and Drnevich equation, Professor Drnevich was consulted. Based on the discussions with Professor Drnevich, it was concluded the k_2 values obtained from the laboratory tests should be multiplied by a factor of about two, resulting in a k_{2max} of 100. The recommended values for use are given in Table 2.5-48. Attachment A3.7 is a letter from Professor Drnevich that is attached for reference and indicates that a k_{2max} of 100 is realistic.

In the soil-structure interaction analysis of the plant structures using soil spring method, a range of soil properties has been used. Q&R Figures 220.15-2 through 220.15-5, extracted from Reference 22, show the upper bound and lower bound soil properties curves used. A comparison of these curves with the values given in Table 2.5-48 shows that the moduli values given in Table 2.5-48 are bounded by the curves shown in these figures. Since the structural evaluation has used the curves shown in the above figures, it is concluded that there is no effect of the revisions in Table 2.5-48 on plant structures. The two earth structures, i.e., the natural slopes surrounding the ultimate heat sink and the submerged dike, have also been reevaluated using the revised moduli values given in Table 2.5-48, and have been found to have adequate factors of safety. (Q&R 220.15)

3.7.1.3 Damping Values

3.7.1.3.1 Critical Damping Values

The damping ratios (expressed as a percentage of critical) used in the analysis of various Seismic Category I structures, systems, and components are listed in Table 3.7-1. These damping ratios conform to Regulatory Guide 1.61.

The damping values used in the analysis of Category I structures are in compliance with the requirements of Regulatory Guide 1.61, Section C.3. Since the maximum combined stress in Category I structures due to static, seismic, and other dynamic loading are not significantly lower than the yield stress and one-half yield stress for SSE and OBE, respectively; Regulatory Guide 1.61, Table 1 damping values were used. (Q&R 220.16) Alternative critical damping values for piping may be used as described in Section 3.7.1.3.2.

3.7.1.3.2 Alternative Critical Damping Values for NSSS Piping

Alternative critical damping values, as provided in ASME Boiler and Pressure Vessel Code, Section III, Division 1 Code Case N-411-1 may be used. When used, the following provisions are applied.

1. The code case damping is applied only to uniform (or envelope) response spectra loading analysis for seismic and seismic-like building filtered hydrodynamic loads and the annulus pressurization loading.

2. The code case damping is applied to a spectral analysis load case in its entirety and is not mixed with other damping values within that one load case.

3. Modal and direction combination of the three earthquake directions are combined in accordance with Regulatory Guide 1.92.

4. Consideration of a sufficient number of modes such that the inclusion of additional modes would not result in more than a 10% increase in response.

5. Assurance that the predicted piping displacements are such that adequate clearance exists with respect to adjacent components and equipment.

6. Line mounted equipment is designed to withstand the increased pipe motion.

7. The code case damping is not applied to piping analytical models that incorporate equipment with natural frequencies below 20 Hertz (Hz).

3.7.1.4 Supporting Media for Seismic Category I Structures

The description of the supporting media for Seismic Category I structures is presented and discussed in Section 2.5. The soil properties used in the design basis seismic analysis (finite element soil model) discussed in answers to Questions 220.14, 220.15, 220.21, and 220.26 are based on the properties given in Table 2.5-48.

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 E-W	N-S	Structural Height Above Grade (ft)
Containment Building	34	199.6	130	(diameter)	191
Auxiliary Building	38.2	195.4	178	122	64
Fuel Handling Building	33.7	200.7	182	151	64
Circulating Water Screen House	44.5	152.9	176	238	32.5
Control Building	43	189.2	219	100	112.2
Diesel Generating and HVAC Building	33	199.2	221.1	106	52
Radwaste Building (substructure only)	43	186.1	232	321	45

* Values obtained using representative parameters for elevation of bedrock. Refer to Figure 2.5-282 and Figure 2.5-373.

3.7.2 Seismic System Analysis

The seismic analysis of the containment and other main building structures is carried out for two building models, one for the single unit building complex presently being constructed, and the second for the two-unit building model. The second unit was planned for construction later on the extension of the single unit basemat. Two separate soil-structure interaction analyses are carried out with the two building models and the corresponding foundation interaction time histories are used for generating various floor spectra in each building model. The envelopes of the floor response spectra from the two analyses are used for the analysis of the equipment and piping supported on various floors.

3.7.2.1 Seismic Analysis Methods

The seismic analysis is performed using the modal superposition method. The member forces and accelerations of mass points are determined by the response spectrum method while the response spectra at various floor elevations for subsystem analysis are generated by the time history method.

All modes with frequencies less than 33 Hz were included in the analysis except when the number of modes required to reach 33 Hz exceeded 30. For these cases, 30 modes were used in the analysis. The following provides the number of modes and the highest frequency considered for the various models:

BUILDING	DIRECTION	NUMBER OF MODES	HIGHEST FREQUENCY
Main Plant	horizontal	25	52.9533
(1 unit)	vertical	26	32.5488
Main Plant	horizontal	25	42.8550
(2 unit)	vertical	27	32.2245
Containment	horizontal	30	27.9994
	vertical	30	26.0777

Note: Unit 2 has been canceled. (Q&R 220.17)

Dynamic modeling of the building structures is described in Subsection 3.7.2.3. The computer program DYNAS (Dynamic Analysis of Structures) is used to analyze the Seismic Category I building structures. The description of this program is given in Appendix C.

Figures 3.7-28 through 3.7-30 are typical sketches of the horizontal seismic model for one and two unit stations. Figure 3.7-79 is the sketch of the horizontal seismic model for the circulating water screen house. Rigid slabs at various floor elevations are connected by shear wall springs. The containment is modeled as a lumped mass spring model. The horizontal models are analyzed for X (E-W direction) and Y (N-S direction) excitations and the results are combined as described in Subsection 3.7.2.6.

As in the horizontal analysis, both response spectrum and time history methods of analysis are performed on the vertical model also using the DYNAS program.

3.7.2.2 Natural Frequencies and Response Loads

3.7.2.2.1 Horizontal Excitation

The periods, mode shapes, and dynamic responses of the structural lumped mass models shown in Figures 3.7-28 through 3.7-30, and 3.7-79 are computed using the program DYNAS. Summaries of the modal frequencies and modal participation factors for the containment model, single unit main plant model, two unit main plant model, and circulating water screen house model are presented in Tables 3.7-2 through 3.7-4, and 3.7-12.

Seismic response loads for the safe shutdown earthquake for the containment wall and major Seismic Category I shear walls are shown in Figures 3.7-31 through 3.7-37.

Response spectra for the design of subsystems consist of envelopes of the responses obtained for the single unit and two unit plant models. Design horizontal acceleration response spectra for the SSE in the East-West and North-South directions at the base slab (elevation 712 feet 0 inch), top of the reactor pedestal (elevation 742 feet 8 inches), drywell floor (elevation 803 feet 3 inches), grade floor (elevation 737 feet 0 inch), and mezzanine floor (elevation 762 feet 0 inch) are shown in Figures 3.7-38 through 3.7-47.

The design horizontal acceleration response spectra for the circulating water screen house in the east-west and north-south directions at the base slab (elevation 653 feet 6 inches), intermediate floor (elevation 682 feet 6 inches), main floor (elevation 699 feet 0 inch) and roof (elevation 730 feet 0 inch) are shown in Figures 3.7-82 through 3.7-89 for a SSE excitation.

3.7.2.2.2 Vertical Excitation

The modal frequencies and participation factors of the containment, main structure, and circulating water screen house lumped mass models shown in Figures 3.7-48, 3.7-49, 3.7-80, and 3.7-81 are presented in Tables 3.7-5 through 3.7-7, and 3.7-13.

Forces in the structures resulting from a vertical excitation are obtained by a response spectrum method of analysis. Seismic response loads in the containment wall for the single unit and two unit lumped mass systems are shown in Figures 3.7-50 and 3.7-51.

Enveloped vertical acceleration response spectra for the main plant at the base slab (elevation 712 feet 0 inch), top of the reactor pedestal (elevation 742 feet 8 inches), drywell floor (elevation 803 feet 3 inches), the grade floor (elevation 737 feet 0 inch), and the mezzanine floor (elevation 762 feet 0 inch) are shown in Figures 3.7-52 through 3.7-56.

The design vertical wall acceleration response spectra for the circulating water screen house at the base slab (elevation 653 feet 6 inches), intermediate floor (elevation 682 feet 6 inches), main floor (elevation 699 feet 0 inch), crane level (elevation 719 feet 0 inch), and roof (elevation 730 feet 0 inch) are shown in Figures 3.7-90 through 3.7-94 for SSE excitation.

3.7.2.3 Procedure Used for Modeling

3.7.2.3.1 Designation of System Versus Subsystem

Analysis of a nuclear power plant complex subjected to seismic excitations is divided into two parts. The first is analysis of "seismic systems" comprising major buildings and structures which house and/or support Seismic Category I systems and components.

The second part is an analysis of "seismic subsystems," which include Seismic Category I systems and components.

Major structures which are analyzed as seismic systems are:

a. main plant complex - control, diesel and HVAC, radwaste and Unit 1 and 2 fuel, auxiliary and turbine buildings;

b. containment wall;

c. containment inner structures - drywell wall, shield wall, weir wall and RPV pedestal; and

d. circulating water screen house.

3.7.2.3.2 Decoupling Criteria for Subsystems

All subsystems such as equipment and piping are decoupled from the floors on which they were supported, since the mass of the structures is large relative to the subsystem masses. However, the masses of these subsystems are included with the structural mass of the supporting floor slabs.

No specific ratios between subsystem mass and system mass, R_m, or between fundamental frequencies of subsystem and system, R_f, used in subsystem decoupling, since no quantitative criteria were available at the time the seismic model was generated. All subsystems, except the RPV mass, were lumped at the appropriate location in the seismic model. The RPV was modeled as part of the seismic model.

The subsystems, with the exception of the RPV, generally have a small mass ratio (R_m is less than 0.01) or frequencies away from resonance with the system (R_f <= 0.8 or R_f >= 1.25). (Q&R 220.18)

3.7.2.3.3 Lumped Mass Consideration

Two independent models are used to obtain responses to horizontal and vertical excitations. Since the response in the horizontal direction due to a vertical excitation (and vice versa) is negligible, the models for horizontal and vertical analyses are decoupled. Horizontal responses are obtained for excitations along two principal horizontal axes. The results of these analysis are then combined using a square-root-of-the-sum-of-the squares (SRSS) method.

For the main building horizontal model, all shear walls and slabs have been modeled at their physical location. This assures the adequacy of the model.

The containment horizontal model consists of 270 degrees of freedom. Thirty modes were extracted from this model for the modal seismic analysis. This meets the SRP requirements that the degrees of freedom be at least two times the number of modes with frequencies less than 33 CPS.

For the main building and containment vertical models, each slab location and each of the slab panel dynamic characteristics is modeled. This assures the adequacy of the model. (Q&R 220.19)

3.7.2.3.3.1 Model for Horizontal Excitation

In the lumped mass idealization, the entire mass of the structure is concentrated at a number of discrete points. In general, each mass point has six degrees of freedom. However, certain degrees of freedom may be neglected depending on the configuration of the structure and the type of excitation. The concrete elements connecting the lumped masses are modeled as linear elastic members.

The main plant complex and circulating water screen house are modeled as a shear structure consisting of rigid concrete slabs interconnected with shear walls. The predominant mode of deformation is shear deformation. Consequently, the only significant rotations are those about the vertical axis, and for each slab only three degrees of freedom are considered, two horizontal translations and rotation about the vertical axis.

Three mass parameters, corresponding to the three degrees of freedom, are associated with each slab. The mass parameters associated with the two horizontal translations are the same and are equal to the mass of the slab. The mass parameter associated with rotation about the vertical axis is equal to the mass polar moment of inertia of the slab about a vertical axis through its centroid. These mass parameters are computed from the mass distribution of the slabs, equipment, piping and tributary walls. Since the shear walls are distributed horizontally, the model slabs are actually treated as rigid bodies with only horizontal dimensions. Shear wall forces are a function of wall stiffness, location of the wall with respect to the centroids of the slabs and the relative motion of the slabs to which the shear walls are attached. Concrete shear walls are treated as deep beams for computation of stiffness, while conventional frame or truss analysis methods are used to compute the stiffness of steel framing.

The containment structures is modeled as a frame structure with all six degrees of freedom considered at the lumped mass locations. The discrete mass at each node includes equipment, piping, wall, and slab masses. The mass of the water in the suppression pool and containment pool is lumped with the containment wall and drywell masses at appropriate elevations. The stiffnesses of the structural elements are computed based on the geometry of the structure and the assumption of linear elastic behavior.

The containment model includes a model of the reactor pressure vessel (RPV) and internals (see Figure 3.7-30). A detailed description of the RPV analysis is included in Subsection 3.7.3.14.

The shear structures and frame models for the main building complex and containment building are coupled for use in the soil-structure interaction analysis described in Subsection 3.7.2.4. After the interaction time history and response spectrum at the foundation have been computed, the models are decoupled and further analysis is performed on each separately.

The sloshing effects of the water in the suppression pool, containment pool, and fuel pools in the fuel handling building were evaluated in accordance with Housner's method explained in Reference 23. It was found that because of the pool geometry, the convective wall pressure was less than the impulsive pressure. In the design of the pools, the impulsive pressure was considered acting throughout the depth of the pool which is conservative. (Q&R 220.20)

3.7.2.3.3.2 Model for Vertical Excitation

A frame system is used to model the main plant complex, the containment building, and the circulating water screen house for analysis of vertical excitations. Since axial deformations are the dominant mode for vertical excitations, only the vertical degrees of freedom are considered in the analysis. Several single-degree-of-freedom systems are connected to the wall system (See Figures 3.7-48, 3.7-49, and 3.7-81) to simulate the multiperiod characteristics of the slab and beam systems.

Masses are concentrated at wall-slab intersections and at the center of the slabs. Wall masses are distributed equally to the adjacent slabs. One-third of the total slab mass, including piping and equipment, is assumed to be effective for slab vibrations. The remaining mass is lumped with the wall at that elevation.

After the vertical interaction time history and response spectrum at the foundation are obtained, further analyses of the containment and main plant are performed separately.

The RPV and internals are included in the containment model (see Figure 3.7-48). Detailed analysis for the RPV is discussed in Subsection 3.7.3.14.

3.7.2.4 Soil-Structure Interaction

The supporting media for various Seismic Category I structures are given in Subsection 3.7.1.4. The horizontal soil-structure interaction analysis is done using a finite element soil model.

Strain-dependent soil parameters used in the interaction analysis are presented in Table 2.5-48. The design time history is applied at the free field ground surface level. The SHAKE program (described in Appendix C) is used to analyze the model shown in Figure 3.7-11 for the main plant and containment building and Figure 3.7-76 for the circulating water screen house in order to obtain the strain-compatible shear modulus and damping values for each layer for both OBE and SSE earthquakes. Corresponding compatible rock motions for OBE and SSE are also obtained from this analysis.

Strain compatible shear modulus (G) and damping values for each layer for horizontal OBE and SSE excitations are presented in Table 3.7-14 for the main plant and containment. Vertical ground motion is considered to travel as a compression wave. In order for "SHAKE" to perform a compression wave analysis, the shear moduli obtained from horizontal excitations are multiplied by a factor $\frac{2(1-\mu),}{(1-2\mu),}$ where μ is Poisson's ratio. (This factor is equal to the ratio of compression wave and shear wave velocities squared.) Damping values used for the vertical analysis are the same as for the horizontal analysis.

The strain levels in Table 3.7-14 are based upon the design time history and are consistent with the design earthquake (0.26g, Regulatory Guide 1.60 spectrum at the ground surface for SSE) and are not considered high.

For soil properties variation, see the response to Question 220.15. (Q&R 220.21)

For the main plant and containment building, a set of soil properties and the corresponding rock motions for both OBE and SSE earthquakes are obtained, then the axisymmetric finite element soil model shown in Figure 3.7-57 is used for extracting normalized modes of soil using the computer program DYNAX (see Appendix C for description of DYNAX program).

In the DYNAX soil model shown in Figure 3.7-57, the structural basemat is modeled through the use of massless rigid shell elements at the foundation elevation. The model has both translational and rotational degrees of freedom at the interface node. The rigid elements at the foundation element simulate the actual distributed interface between the structural basemat and the soil, even though the structure is connected at only one point in the mathematical model.

The effects of neglecting the interaction between the walls and the soil is insignificant because:

i) The embedment depth (d) is small compared to the structural base dimension (B). For the one unit model, the average d/B ratio is 0.07. For such small embedment ratios, the embedment effects on soil impedance are negligible.

ii) The soil-structure interaction responses using perfect contact between the foundation and the side soil may not be realistic, because a partial loss of contact between the soil and the foundation leads to a pronounced reduction in rocking stiffness.

iii) Neglecting the effect of side soil leads to more rocking of the structure, and thus is conservative.

It should be noted that even though the interaction between the side soil and the walls is neglected for the soil-structure interaction analysis, the walls are designed to resist dynamic earth pressure loads expected during the seismic event. (Q&R 220.22)

The three dimensional axisymmetric soil media model is shown in Figure 3.7-57. In this model the boundary condition are as follows:

(i) All surface nodes are free to model zero traction condition at the surface;

(ii) The soil basemat interfact nodes are connected by rigid elements to simulate the rigid foundation interface;

(iii) Nodes at the bottom boundary are fixed. This boundary coincides with the physical rock-soil interface at the CPS site.

(iv) The lateral free field boundary nodes are free to move in the horizontal direction but are contained in the vertical direction. This boundary condition simulates the shear beam behavior for the soil media, consistent with the concept that horizontal seismic motions can be modeled by vertically propagating shear waves. To assure that the imposed lateral boundary condition has no practical effect on the soil structure interaction responses, the lateral boundary was chosen to be sufficiently far from the structure.

To verify that the lateral boundary location is far enough from the structures in the coupled soil structure interaction model, response at the foundation elevation at the lateral boundary was computed in the coupled soil structure interaction model. This response was compared to the foundation elevation free field response obtained from the SHAKE analysis. This comparison is presented in Figure 3.7-95. It can be observed that the soil model for SSI is large and does simulate the free field conditions at distances sufficiently far from the structure (Q&R 220.24)

Using the modal synthesis technique, the three dimensional building models (one and two unit) shown in Figures 3.7-58 and 3.7-59 are analyzed in the two orthogonal horizontal directions X and Y using the program DYNAS for OBE and SSE earthquakes. One discrete torsional soil spring and corresponding mass are included to account for possible torsional interaction due to the non-symmetric nature of the building complex. The torsional spring constant for each soil layer is calculated as (Reference 13):

$$k = \frac{16}{3} G r_o 3$$

where r_o is the radius of the effective area. For a rectangular base with dimensions B and L,

$$r_o = \sqrt[4]{\frac{BL\left(B^2 + L^2\right)}{6\pi}}$$

G = shear modulus of soil layer obtained from SHAKE.

Finally, the total spring constant is calculated by adding the torsional stiffnesses of springs in series for all layers. The effective mass inertia (I) of the soil participating in torsional vibration is taken as (Reference 13):

$$I = \frac{0.3\lambda r^5}{g^o}$$

where λ = unit weight of soil.

The modal damping values required in DYNAS are obtained from the DYNAX program by using the strain-compatible damping values obtained from SHAKE for each layer.

The DYNAX program computes the mixed modal damping for the modal synthesis soil-structure interaction analysis option as follows:

a. Various layers in the DYNAX soil model (Figure 3.7-57) are assigned damping values consistent with those obtained in the SHAKE analysis.

b. The mixed modal damping is generated using Equation 3.7-3. (Q&R 220.23)

The resulting interaction spectra in the two orthogonal X and Y directions at the foundation of the structures for one and two unit building models are obtained for both OBE and SSE earthquakes. These may be found in Calculation SDQ51-14AS02. The horizontal design response spectra at relevant locations of the structure are generated in the two models using a fixed base model subjected to the corresponding interaction time history. This decoupled fixed base analysis is justified as the foundation torsion, and rocking is found to be insignificant. Forces in the shear walls are generated by subjecting the two fixed-base building models to the corresponding 15% widened interaction response spectrum. Shear walls are finally designed for the maximum of the two unit building models.

Two-dimensional finite element models shown in Figures 3.7-77 and 3.7-78 were used for the circulating water screen house for the coupled soil-structure interaction analysis. The computer program LUSH (described in Appendix C) was used for this analysis.

For excitation in the vertical direction, a lumped mass multiple spring model as shown in Figure 3.7-68 is used instead of the finite element soil model. The portion of soil from foundation to rock level is modeled as a prismatic column of soil equal in area to the area of the basemat. The layering scheme of the soil is the same as that used for horizontal soil-structure interaction analysis. The Young's modulus E for the soil layers in the vertical direction is calculated from the strain compatible shear modulus values obtained from the horizontal SHAKE analysis using the relation

$$E = 2G\left(1 + \mu\right)$$

where μ = Poisson's ratio.

The SHAKE program is used to obtain a compatible rock motion for both OBE and SSE earthquakes (using the program to solve the problem of compression wave propagation instead of shear wave) and specifying the design vertical time history at the free field surface level.

Each layer of soil is then represented by an axial spring with its mass lumped at its two ends. The stiffness of each of these axial springs is computed as (AE/L) where A is the surface area of the layer (equal to the building basement area) and L is the thickness of that layer. The above soil model, together with the vertical building models (one and two unit) described in Figures 3.7-69 and 3.7-70 are analyzed for OBE and SSE earthquakes using the program DYNAS. The interaction spectra at the foundation of the building are generated by subjecting the coupled soil structure model to the compatible rock motion. These are shown in Figures 3.7-71 through 3.7-74. The vertical design spectra are at relevant locations of the structure are generated for the two models, using a fixed-base model subjected to the corresponding vertical interaction time history.

All the power block structures in CPS are supported by a single common basemat and interconnected by shear walls in both the north-south and east-west directions. This structural configuration results in a small height-to-base dimension and also leads to a large torsional moment of inertia. This leads to relatively small rocking and torsional motions. To assure that the torsional and rocking motions are insignificant, thus justifying the use of a decoupled fixed base model, the response obtained from the coupled soil-structure interaction model were compared to those obtained from the uncoupled fixed base model. Typical comparison of the coupled vs. uncoupled model response are provided in Figures 3.7-96 (el. 781 feet 0 inch), 3.7-97 (el. 825 feet 0 inch), and 3.7-98 (el. 874 feet 0 inch). The locations of these slabs are shown in Figure 3.7-29. It can be observed that the two models lead to approximately the same response, thus justifying the use of the decoupled fixed base model. The CPS structural layout is such that the horizontal excitation leads primarily to shear and bending in shear walls, columns, and the containment. The vertical excitation leads to axial loads in shear walls and columns and bending of floor slab panels. In the absence of any significant rocking, as shown above, the vertical and horizontal responses are uncoupled, and thus can be computed using separate horizontal and vertical models. For the CPS project, separate vertical and horizontal models are thus used. It should be noted that even though the horizontal and vertical excitation responses are obtained using separate models, all structures, piping, and equipment are designed for the combined effects of the three components of earthquake. (Q&R 220.25)

3.7.2.5 Development of Floor Response Spectra

3.7.2.5.1 Introduction

If a structure is subjected to an earthquake, the base of a subsystem (or equipment) mounted on a 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.

Final design response spectra are obtained by enveloping the response spectra from the analyses of the single unit and two unit systems.

3.7.2.5.2 Horizontal Response Spectra

Time history analyses of each building system are performed on the horizontal seismic models as discussed in Subsection 3.7.2.1. The following general procedure was used to develop the horizontal seismic subsystem input:

a. The responses at each slab of interest were obtained by exciting the structure separately along the two principal axes (X and Y) of the structure. The responses obtained from the two components are combined on a square-root-of-the sum-of-the-squares basis (SRSS).

   The justification of decoupling vertical and horizontal models is provided in response to Question 220.25. It should be noted that even though the horizontal and vertical excitation responses are obtained using separate models, all structures, piping, and equipment are designed for the combined effects of the three earthquake components. (Q&R 220.27)

b. Response spectra are generated at 0.5, 1, 2, 3 and 4% of critical damping for the OBE and 1, 3, 4, 7, and 10% of critical damping in the main plant and 1, 2, 3, 4, 7 and 10% of critical damping in the containment building for the SSE.

c. Response spectra are generated at each slab which supports Seismic Category I subsystems or components. Fifty periods from 0.02 to 2.0 seconds are used to define each spectrum curve.

   The periods used to generate response spectra at each slab are shown in the attached Table 3.7-15. As can be seen from the table, the periods selected are at very fine intervals and included the natural period of most of the supporting structures. In cases where the period selected for spectrum generation differed from the natural period of the supporting structure, the deviation is negligible (the maximum deviation between the spectral period and the natural period of the supporting structure is only 0.01 second). (Q&R 220.28)

d. For the design of subsystems, the peaks of the response spectra are widened by 15% to either side of the peak.

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 determining responses in the horizontal direction. Response spectra are generated for uncoupled time history motion in the vertical direction at slabs and discrete mass points at the wall/slab junction. For the design of subsystems, the response spectra are widened by 15% to either side of the peak.

3.7.2.6 Three Components of Earthquake Motion

Seismic responses resulting from analysis of systems due to three components of earthquake motion are combined in the following manner as per Regulatory Guide 1.92:

$$R = \sqrt{R_x^{\ 2} + R_y^{\ 2} + R_z^{\ 2}} \qquad (3.7\text{-}1)$$

where:

R = design seismic response.

R_x, R_y, and R_z are probable maximum, codirectional 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

3.7.2.7.1 Systems Other Than NSSS

When a response spectrum method of analysis is used to analyze a system, the maximum response (accelerations, shears, and moments) in each mode is calculated independent of time, whereas actual modal responses are independently time dependent and maximum responses in different modes do not occur simultaneously. Based on Regulatory Guide 1.92 and References 5 and 8, the final response R is computed as:

$$R = \left[\sum_{K=1}^{N} \sum_{1=1}^{N} \left| Rk\ R1\, k1 \right| \right]^{1/2} \qquad (3.7\text{-}2)$$

where:

$$\varepsilon k_1 = \left\{ 1 + \left[\frac{\omega' k - \omega' 1}{\beta' k \omega k + \beta' 1 w 1} \right]^2 \right\}^{-1}$$

in which:

$$\omega' k = \omega k \, [1 - (\beta'_k)^2]1/2$$

$$\beta' k = \beta k + \frac{2}{t_d \omega_k}$$

where ω_k and βk are the modal frequency and damping in the kth mode, respectively and t_d is the duration of the earthquake. For the time history method of seismic analysis, the displacements, acceleration, shears, and moments due to each mode are added algebraically at each instant of time to obtain the final response.

3.7.2.7.2 NSSS

In a response spectrum modal dynamic analysis, if the modes are not closely spaced (i.e., if the frequencies differ from each other by more than 10 percent of the lower frequency), the modal responses are combined by the square root of the sum of the squares (SRSS) method as described in Subsection 3.7.2.7.2.1. If some or all of the modes are closely spaced, a double sum method, as described in Subsection 3.7.2.7.2.2, is used to evaluate the combined response. In a time-history method of dynamic analysis, the algebraic sum at every step is used to calculate the combined response. The use of the time-history analysis method precludes the need to consider closely spaced modes.

3.7.2.7.2.1 Square Root of the Sum of the Squares Method

Mathametically, this SRSS method is expressed as follows:

$$R = \left(\sum_{i=1}^{n} (R_i)^2 \right)^{1/2}$$

where:

R = Combined Response

R_i = Response in the i^{th} mode

n = Number of Modes considered in the analysis.

3.7.2.7.2.2 Double Sum Method

This method is defined mathematically as:

$$R = \left(\sum_{k=1}^{N} \sum_{s=1}^{N} \left| R_k \; R_s \right| \varepsilon ks \right)^{1/2}$$

where:

R = Representative maximum value of a particular response of a given element to a given component of excitation

R_k = Peak value of the response of the element due to the k^{th} mode

N = Number of significant modes considered in the modal response combination

Rs = Peak value of the response of the element attributed to s^{th} mode

where:

$$\varepsilon ks = \left[1 + \left\{ \frac{\omega' k - \omega' s}{\left(\beta'_k \; \omega_k \; + \beta'_s \; \omega_s \right)} \right\}^2 \right]^{-1}$$

in which:

$$\omega' k = \omega k \left[1 - \beta_k^2 \right]^{1/2}$$

$$\beta' k = k + \frac{2}{t_d w_k}$$

where ω_k and β_k are the modal frequency and the damping ratio in the kth mode, respectively, and t_d is the duration of the earthquake.

Subsection 3.7.2.7.1 describes the double sum expressions used by Sargent & Lundy to combine modal responses, whereas Subsection 3.7.2.7.2.2 describes the double sum expressions used by General Electric. The differences in these Subsections are minor and of no engineering significance. The following paragraphs summarize the differences and evaluate their significance:

a. For the second summation, index variable i is used in Subsection 3.7.2.7.1, and s is used in Subsection 3.7.2.7.2.2. This difference is of no significance because the response R is not dependent on the summation index.

b. The coupling term $\varepsilon_{\kappa}\ell$ (or ε_{ks}) appears within the absolute sign in Subsection 3.7.2.7.1 and outside the absolute sign in Subsection 3.7.2.7.2.2. The coupling term $\varepsilon_{\kappa}\ell$ (or ε_{ks}) is always a positive quantity; thus it does not matter whether e_k is outside or inside the absolute sign. Both equations will lead to the identical response R.

c. In defining the term ω_k Subsection 3.7.2.7.1 uses βk, whereas Subsection 3.7.2.7.2.2 uses β'_k. It is debatable whether β'_k or β_k should be used in computing ω'_k; in S&L's opinion the use of β'k in the equation is correct. However, the concern over the use of β_k or β'_k is academic and does not affect the seismic design, because the damping values used in the CPS design are small (1% to 7%). For these small damping values, the variance in design responses computed using β_k or β'_k is of the order of 0.1% and of no engineering significance. (Q&R 220.29)

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 ensure 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

To account for the expected variation in structural properties, damping and soil properties, the peaks of various floor response spectra curves are widened by 15% on the period scale to either side of the peak for horizontal as well as vertical components, as per Regulatory Guide 1.122.

3.7.2.10 Use of Constant Vertical Static Factors

The Seismic Category I structures, systems, and components are analyzed in the vertical direction using the methods described in Subsection 3.7.2.1. However, beams in a floor slab are designed using a constant vertical acceleration equal to 1.5 times the acceleration value corresponding to the fundamental frequency of the beam from the applicable wall response spectrum (see Reference 14).

Subsection 3.7.2.10 is in conformance with the provisions of the SRP. There is no deviation between the analysis and design method stipulated in the FSAR and SRP Section 3.7.2-II.1.b, which permits the use of any rational and justifiable equivalent static load method. Justification is given below.

1. SRP Section 3.7.2-II.1.b(3) applies only to the design of floor attached structures, equipment, and components, and is based on a static load method which involves no analysis, i.e., no frequency calculation or modeling of the component.

2. The equivalent static load design method stated in Subsection 3.7.2.10 for design of floor framing is a more comprehensive and realistic method. It involves modeling each main floor framing member and determination of the fundamental frequency of the member, consideration of the source of seismic excitation, and includes the effect of higher mode participation. The adequacy and conservatism of this method has been evaluated by comparing the results with a dynamic analysis for a typical floor framing. The results of this analysis were published in the Proceedings of the ASCE Spring Convention in Dallas, Texas in April 1979 (FSAR Reference 14, Subsection 3.7.5).

3. The justification for using the wall response spectrum rather than the floor response spectrum is as follows:

   The floor framing members are supported by steel columns. In the vertical seismic model, the columns are included with the walls. The seismic response of the floor framing members is given by the response spectra of the supporting columns. The column response spectrum is given by applicable wall spectrum. Therefore, it is appropriate to use the wall response spectra for the design of the floor framing members.

4. Since the floor framing member is modeled as a single degree of freedom system, an amplification factor of 1.5 is used to account for higher mode participation. This factor is a conservative value. The behavior of a typical steel floor framing member is close to that of a single degree of freedom system, in which higher mode participation is insignificant. The use of 1.5 as the amplification factor for flexible beams with frequencies lower than 33 Hz will ensure the design adequacy of the equivalent static load method used herein. (Q&R 220.30)

3.7.2.11 Method Used to Account for Torsional Effects

The complex building structures with heavy equipment and concrete slabs at the various floor elevations have asymmetrical mass-stiffness distribution. Consequently, the slab rotation about the vertical axis occurs when this type of structure is subjected to lateral loads. This torsional effect in slabs is accounted for by including a torsional degree of freedom in each slab of the horizontal building structure model.

In the CPS structural design, the additional 5% accidental torsion was not included because this requirement did not exist when the plant was designed and constructed. However, the CPS structures have been designed to resist large torsional loads due to the asymmetry of the equipment and structural layout. Thus, the additional accidental torsion equal to 5% of the dimension does not result in significant additional forces in the lateral load resisting structural elements (shear walls). This conclusion is based on a CPS-unique analysis where the SSE design shear wall forces were compared to those resulting from the consideration of accidental torsion as required by the NRC staff. This analysis shows that the shear wall load increased an average of 7% with a range of 0 to 15%. Typical values of these increases are presented in Table 3.7-16. The small magnitude of the additional loads leads us to conclude that the effect of the 5% accidental torsion on the CPS design is insignificant. (Q&R 220.31)

3.7.2.12 Comparison of Responses

The time history method of analysis is used to generate acceleration response spectra at the lumped mass locations for the design of the structural subsystems and piping components.

The response spectrum method of analysis is used to generate forces and moments for the design of structural components. A comparison of the forces and moments generated by the response spectrum and the time history methods of analysis for the single unit containment wall due to an east-west SSE excitation is given in Table 3.7-8.

For the CPS design, a response spectrum analysis was performed to compute structural forces and moments. A time history analysis was performed to obtain floor response spectra and inter-story drift. No structural forces and moments were computed from the time history method; thus, a comparison of time history and response spectra analysis procedures used on the CPS project are consistent with Regulatory Guide 1.92 requirements. Given the acceptability within the engineering profession of using the time history and response spectra methods for seismic analysis, we do not believe a comparison of forces obtained for the CPS project using the two methods should be required.

The lower acceleration values for slabs close to the support obtained with the response spectrum method are to be expected because the acceleration values computed in the response spectrum method are pseudo-absolute values obtained by multiplying the modal displacements (relative to the supports) by the square of the modal frequency, ω^2. In the time history method, the acceleration values are the absolute values and are obtained by adding the base acceleration to the relative acceleration values computed in the modal method. This condition arises because responses in the modal seismic analysis are expressed by fixed base mode shapes. Because structural forces and moments are a function of the relative displacement of the node points, the response spectra and time history methods lead to very close forces. Reference 2 of Regulatory Guide 1.92 (Reference 8) presents a comparison of the forces obtained from the time history and the response spectra methods using various modal combination rules and a wide range of structural frequencies. Note that with the double sum method (the method used in design), a close comparison between the time history and the response spectrum method results is achieved for a wide range of structural configurations. (Q&R 220.32)

3.7.2.13 Method for Seismic Analysis of Dams

The method of analysis used for evaluating the seismic stability of Seismic Category I dams is described in Subsection 2.5.5.2.4.

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 by the double sum method to determine the probable maximum overturning moment in the structure.

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:

$$\beta j = \frac{\sum_{i=1}^{n} (\Phi_j)^T \beta_i (M)^i (\Phi_j)}{(\Phi_j)^T (M)(\Phi_j)} \qquad (3.7\text{-}3)$$

$$\beta j = \frac{\sum_{i=1}^{n} (\Phi_j)^T \beta_i (K)^i (\Phi_j)}{(\Phi_j)^T (K)(\Phi_j)} \qquad (3.7\text{-}4)$$

where:

n = total number of components,

β_j = composite modal damping for mode j,

β_i = critical modal damping associated with component i,

$[\Phi_j]_i$ = mode shape vector corresponding to element i, region and mode j,

$[M]_i$, $[K]_i$ = subregion of mass or stiffness matrix associated with component i,

[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 3.7-3 and 3.7-4 would give identical results. In 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 3.7-4 is used in the analysis of fixed-base dynamic models.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Seismic Analysis Methods

3.7.3.1.1 Seismic Analysis Methods for Piping

Each pipeline is idealized as a mathematical model consisting of lumped masses connected by elastic members. This means that the weight properties of the subsystem have been lumped on discrete joints and the members connecting these joints are assumed to have all geometric and elastic properties but no weight.

The piping subsystem is treated as a space frame having six degrees of freedom for each joint (three translations and three rotations). The displacements of a joint in space can be defined by the above-mentioned six degrees of freedom. The stiffness matrix of 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 either the time-history or the response spectrum method of analysis. With the response spectrum method of analysis, when the piping subsystem is anchored and/or supported at points with different excitations, the analysis is performed using the enveloped response spectra of all response spectra which apply.

3.7.3.1.1.1 Modal Method of Analysis

The modal method of analysis is divided into three basis steps:

a. Generation of stiffness and mass matrices for the complete piping subsystem from its geometrical mechanical property.

b. Formulation and solution of the eigenvalue problem to get eigenvalues and eigenvectors (the eigenvalues are the frequencies, and the eigenvectors are the mode shapes of natural vibration of the system).

c. Solution of uncoupled equations for response due to the specified excitation.

3.7.3.1.1.2 Stiffness Matrix Generation

The stiffness matrix of the piping subsystem depends only on the geometric and elastic properties of the piping subsystems. The joint stiffness matrix is obtained by summing the individual member stiffness matrices connected to that particular joint.

3.7.3.1.1.3 Mass Matrix Generation

The mass matrix for a piping subsystem modeled as a lumped mass system with N degrees of freedom is a diagonal matrix of size N x N.

The weight components at the joints are computed from the given weight properties of the members connected to the particular joints.

3.7.3.1.1.4 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 analysis is used to obtain them.

The analysis of piping subsystems due to relative seismic support motions consists of two phases. In the first phase, the structural time history responses are generated and reduced to a format suitable for piping analysis; in the second phase, the piping responses due to these structural movements are computed. The details of the various steps are as follows:

a. The structural building model is analyzed using the DYNAS Program (Appendix C) to obtain three sets of floor displacement time histories, one each for the two horizontal and vertical excitations. These displacement time histories typically have 1,000 discrete points. For each excitation, a selection of floor responses at 100 random time instances for each of the time history sets is done. These random selections are stored in computer files for use in the piping analysis.

b. For each excitation, the piping subsystem is analyzed for the 100 relative support displacements obtained in a. above; the responses of that analysis are enveloped and then the enveloped responses are multiplied by 1.3.

c. The responses from each excitation are combined by the square root of the sum of the squares method (SRSS).

d. The piping secondary stresses are evaluated using the OBE relative displacement. However, in evaluating the maximum support loads, both the OBE and SSE relative displacements are considered as primary loads in support design at the specified service level.

The above procedure yields a reasonable estimate of the maximum responses and support reactions due to seismic relative support movements.

3.7.3.1.2 Seismic Analysis Methods for Equipment

The qualification procedure for equipment is discussed in Section 3.9.2.2.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 BOP Piping

Five occurrences of Operating Basis Earthquake (OBE) Loadings are assumed for piping fatigue analysis. Each occurrence of an OBE loading event results in 10 equivalent maximum stress cycles (ASME B&PV Code, Section III, Appendix N-1214). A total of 50 maximum stress cycles are used.

3.7.3.2.2 BOP Equipment

For those pieces of equipment where testing is an acceptable method of qualification, the test duration shall simulate the effect of five OBEs followed by one SSE with each test duration at least equivalent to the strong motion component of the earthquake or a minimum of 10 seconds. The approach used is recommended by IEEE 344. For all other BOP equipment, the qualification method is discussed in Section 3.9.

3.7.3.2.3 NSSS Piping and Component

3.7.3.2.3.1 NSSS Piping

Sixty-peak OBE cycles are postulated for fatigue evaluation.

3.7.3.2.3.2 Other NSSS Equipment and Components

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 N 69° W component, 30 seconds; and March 1957, Golden Gate 80 component, 13.2 seconds. The modal response was truncated such that the response of three different frequency bandwidths could be studied, 0+-10 Hz, 10-20 Hz, and 20-50 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 3.7-11 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.

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 from Table 3.7-11 according to the frequency range within which the fundamental frequency lies.

c. For fatigue evaluation, 1/2% (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 for many plants show 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. To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, 10 peak stress cycles are postulated for fatigue evaluation.

Subsection 3.9.1.1 presents the number of fatigue cycles used in the design of GE-supplied subsystems.

3.7.3.3 Procedure Used for Modeling

3.7.3.3.1 Modeling of the Piping System

3.7.3.3.1.1 Modeling of the Piping System for BOP Systems

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. Concentrated weights on the piping system, such as motor operated valves are modeled as lumped masses. The torsional effects of the valve operators or other equipment with offset center of gravity with respect to the centerline of the pipe are included in the analytical model.

3.7.3.3.1.2 Modeling of NSSS Piping Systems

3.7.3.3.1.2.1 Modeling of Piping Systems

The continuous piping system is modeled as an assemblage of three dimensional straight or curved pipe elements. The mass of each pipe element 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. In addition, mass points are located at all points on the piping system where concentrated weight such as valves, motors, etc. are located and also at points of significant change in the geometry of the system. 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 center line of the pipe is included in the analytical model. If the torisional effect is expected to cause pipe stresses less than 500 psi, this effect may be neglected.

The criteria employed for decoupling the main steam and recirculation piping systems for establishing the analytical models to perform seismic analysis is given below:

a. The small branch lines (6-in. diameter and less) are decoupled from the main steam and recirculation piping systems and analyzed separately because the dynamic interaction is insignificant due to the disparity in the masses of the two lines.

b. The stiffness of all the anchors and its supporting steel is large enough to effectively decouple the piping on either side of the anchor for analytic and code jurisdictional boundary purposes. The RPV is very stiff compared to the piping system and thus during normal operating conditions the RPV is also assumed to act as an anchor. Penetration assemblies (head fittings) are also very stiff compared to the piping system and are assumed to act as an anchor. The stiffness matrix at the attachment location of the process pipe (i.e., main steam, RCIC, RHR supply or RHR return) head fitting is sufficiently high to decouple the penetration assembly from the process pipe. GE analysis indicates that a satisfactory minimum stiffness for this attachment point is equal to the stiffness in bending and torsion of a cantilever equal to a pipe section of the same size as the process pipe and equal in length to three times the process pipe outer diameter.

3.7.3.3.1.3 Modeling of NSSS Equipment

For dynamic analysis, seismic Category I equipment is represented by lumped mass systems which consist of discrete masses connected by weightless springs. The criteria used to lump masses are:

a. The number of modes of a dynamic system is controlled by the number of masses used. Therefore, the number of masses is chosen so that all significant modes are included. The modes are considered as significant if the correseponding natural frequencies are less than 33 Hz and the stress 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.

3.7.3.3.2 Field Location of Supports and Restraints

In Seismic Category I buildings, only non-Seismic Category I piping has field located supports and restraints. Field locating is done by the contractor in accordance with the design specifications. A walkdown of this piping will ensure that the pipe supports and restraints are installed in accordance with the design specifications and drawings.

3.7.3.4 Basis for Selection of Frequencies

3.7.3.4.1 Introduction-Frequency Range

For seismic response the frequency range of interest is approximately in the range from 1 to 33 Hz.

3.7.3.4.2 Significant Dynamic Response Modes

All modes within a frequency range of interest are included in the dynamic analysis. Generally, the number of modes which are to be considered for the analysis of any given subsystem is dependent on the subsystem characteristics and the amplitude/ frequency content of the input forcing functions. The criterion is to choose the number of modes to cover the peak responses of the applicable loads as much as possible to totally represent the actual piping subsystem responses at the peak response frequency ranges.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

Static Analysis: If it can be shown that the fundamental natural frequency of the equipment is equal to, or higher than, 33 Hz, a static analysis shall be performed to determine the stresses and deflections due to seismic loads. In this case, the seismic forces shall be determined by multiplying the mass of the subassembly or part of the equipment times the maximum floor seismic acceleration at the base of the equipment (zero period acceleration from the response spectra). These forces shall be applied through the center of gravity of the subassembly or the part of the equipment. The stresses resulting from each force (in each of the three directions) shall be combined by taking the square root of the sum of the squares (SRSS) to yield the seismic stresses. The seismic deflections (deflections due to seismic loads) shall be calculated in the same way. These seismic stresses and deflections shall be added to all stresses and deflections resulting from all applicable loads, to obtain the final resultant stresses and deflections, which shall be compared with the design limits.

Simplified Dynamic Analysis: A simplified dynamic analysis may be performed, for flexible equipment, applying the same method as the static analysis but using different values for the accelerations. The accelerations to be used shall be obtained by multiplying the g values corresponding to the fundamental natural frequency from the appropriate response spectra curves by 1.5. If the fundamental natural frequency is not known, a static analysis using 1.5 times the maximum peak of the applicable floor response spectra, as applied to seismic accelerations, is acceptable. The 1.5 factor will conservatively account for possible participation of higher modes. After this, the analysis will follow the same procedure described for static analysis.

Detailed Dynamic Analysis: When acceptable justification for static analysis cannot be provided, a dynamic analysis shall be required, and unless a conservative factor is used to account for the participation of higher modes, a detailed dynamic analysis shall be performed. A mathematical model may be constructed to represent the dynamic behavior of the equipment. The model can be analyzed using the response spectrum modal analysis or time-history (modal or step-by-step) analysis. The maximum inertia forces, at each mass point, from each mode, shall be applied at that point, to calculate the modal stresses and modal deflections. The various modal contributions shall be combined by taking the square root of the sum of the squares of the individual modal stresses or deflections. The case of closely spaced modes shall be defined and considered as stated in NRC Regulatory Guide 1.92. The stresses and deflections resulting from each of the three directions shall be combined by taking the square root of the sum of the squares to obtain the seismic stresses and deflections. These seismic stresses and deflections shall be added to all stresses and deflections resulting from all applicable loads and then compared with the design limits. CPS is in compliance with Regulatory Guide 1.92 in regard to closely spaced modes (See Subsections 3.7.2.7 and 3.7.3.7).

3.7.3.6 Three Components of Earthquake Motion

Seismic responses resulting from analysis of components 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 Procedure for Combining Modal Responses

The method for combining modal responses for systems or subsystems is discussed in Subsection 3.7.2.7.

3.7.3.8 Analytical Procedures for Piping Systems

3.7.3.8.1 Introduction

All Seismic Category I piping is seismically analyzed by either a simplified analysis or a multidegree dynamic analysis, depending on its quality group and nominal size.

3.7.3.8.2 Input Criteria

Seismic responses resulting from analysis of systems due to three components of earthquake motion are combined in the following manner as per Regulatory Guide 1.92:

$$R = \sqrt{R_x^2 + R_y^2 + R_z^2} \qquad (3.7\text{-}11)$$

where:

R = design seismic response.

R_x, R_y, and R_z are probable maximum, codirectional seismic responses of interest (strain, displacement, stress, moment, shear, etc.) due to earthquake excitations in x, y, and z directions, respectively. In cases where more than one response spectrum may be applied to a subsystem, e.g., if the system is supported from locations in the structure having different response spectra, the response spectra used in the analysis of the subsystem will be an envelope of the applicable response spectra.

The total seismic loading obtained from the subsystem analysis consists of two parts, the inertial loading and the loading due to differential anchor movement.

Determination of the applicable seismic loading (moment range) depends on the stress being checked (per Section III of ASME B&PV code). For example, when analyzing Class 1 piping, one-half of the moment range due to inertial effects only is used when satisfying Equation 9 of NB-3652, and the total moment range due to both inertial and anchor movements is used when calculating for S_n and S_p of Equations 10 and 11 of NB-3653.

Similarly for Class 2 and 3 piping stress analysis, the choice of the seismic loading and constituents (inertial or inertial plus differential anchor movements) is determined by the applicable loading condition and the equations being used as per Subsection NC-3652 of Section III.

The loading range due to differential anchor movements is obtained by performing a static analysis of the affected subsystem with the anchor movements acting on each corresponding terminal end. The anchor movements are determined from the seismic analysis of structural systems. See Section 3.7.3.1.1.4.

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 sufficient 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 changes in stiffness due to curved members. Next, the frequencies and mode shapes for all the significant modes of vibration 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 horizontal and vertical displacement and acceleration responses are calculated. The resultant displacement and acceleration responses are determined by combining the maximum response for each mode using the square root of the absolute double sum method. The responses are calculated for each of the three orthogonal directions. Finally, the inertial forces for each direction of earthquake motion for each mode are determined. The stresses due to the inertial forces are determined using the square root of the absolute double sum of the moments for each mode. Horizontal and vertical earthquake excitations are assumed to occur simultaneously. Calculations outlined in this subsection are performed using the PIPSYS computer program for the analysis of a 3-dimensional piping system. A more detailed explanation of the method used to combine modal responses is provided in Section 3.7.2.7.1.

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 in a static analysis to determine the additional stresses due to relative anchor-point displacements.

3.7.3.8.4 Allowable Stresses

Allowables for stresses in the piping caused by an earthquake are in accordance with Section III of the ASME B&PV code. Allowables for stresses in the earthquake restraint components such as shock suppressors are in accordance with the allowable stress limits that may have been established by ASME Section III for B&PV Code at the time the restraint components are purchased.

3.7.3.8.5 Amplified Seismic Responses

The two horizontal and one vertical response spectrum curves are derived for all floor elevations. These curves are used in the design of the piping and its components.

3.7.3.8.6 Use of Simplified Dynamic Analysis

For Seismic Category I non-Class 1 systems, 2-inch diameter and smaller, and Class 1 systems l-inch diameter, which are located in Seismic Category I buildings, a simplified dynamic analysis may be used. This spectra includes OBE with 1% critical damping in accordance with Regulatory Guide 1.61. In order to obtain the emergency condition loads, the upset condition loads are always multiplied by minimum factor of 1.5. (Q&R MEB (DSER) 48)

This method yields only response due to inertial effects. The effects of dynamic end displacement must be considered separately. In this method, piping spans between rigid supports and/ or restraints are treated as independent, simply supported beams. No restraint credit is taken for hangers or restraints not offering stiffness in the direction of the seismic excitation.

The span period, maximum midspan deflection, allowable midspan deflection, and end restraint forces are determined for a given span length. The maximum midspan deflection and restraint forces are a function of the floor response spectrum of the building structure in the vicinity of the piping. The spectra used are for the OBE with 1% critical damping as per Regulatory Guide 1.61. In the cases where the other dynamic loads affect the piping, their responses are also considered as identified by individual building elevation and associated response spectra.

The data described previously are used to:

a. assure that seismic stresses, in conjunction with other primary and secondary stresses, are not greater than the allowable, per ASME Section III, Subsection NC/ND-3600;

b. assure that seismic deflections are not large enough to cause contact between pipe and surroundings; and

c. provide seismic restraint design loads.

3.7.3.8.7 Modal Period Variation

The modal period variation has been considered in the derivation of the response spectrum curves by widening the peaks of those curves (Subsection 3.7.2.9).

3.7.3.8.8 Piping Outside the Containment Structure

Seismic Category I piping located outside the containment, but not buried, is analyzed for seismic effect and differential seismic movement at support points, containment penetrations and at any entry points into other structures, as specified in Subsections 3.7.3.8.1 through 3.7.3.8.7.

3.7.3.8.9 Seismic Category I Subsystem Equipment and Components

The methods of analysis performed for Seismic Category I equipment and components are presented in Subsection 3.9.2.2.

3.7.3.9 Multiple Supported Components With Distinct Inputs

When the component is supported at points with different elevations, the envelope of each applicable elevation response spectrum is developed and conservatively used for the seismic qualification of the component.

The criteria used for considering the piping response due to relative seismic support motions is discussed in Subsection 3.7.3.1.1.4.

BOP equipment, if supported at multiple and different locations, is analyzed to the upper bound of the envelope of the individual response spectra. In addition, the effect of relative support displacements, if applicable, is considered. The responses due to inertia effect and relative displacements are combined by the square root of the sum of the squares (SRSS) method. (Q&R 220.35)

3.7.3.10 Use of Constant Vertical Static Factors

In general, Seismic Category I piping systems are analyzed in the vertical direction using the methods specified in Subsection 3.7.3.1.1, and Seismic Category I equipment is analyzed in the vertical direction using the methods specified in Subsection 3.7.3.5. Vertical static factors used for equipment are discussed in Subsection 3.7.3.5.

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 component 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.

For valve/operator assemblies with natural frequencies greater than or equal to 33 Hz, a simplified (though sufficiently adequate) valve/operator assembly model is considered in the piping analysis to account for eccentricities, thus accounting for bending and torsional effects. Sargent & Lundy has performed a generic study. Representative piping systems were considered using a detailed finite element representation of the valve assembly to account for its flexibility. The results of this study were compared with similar cases where valves were modeled as rigid in the piping analysis. Amplification factors resulting from this comparison will be used to evaluate any flexible valves. Sargent & Lundy will use the results of this study to qualify the flexible valves. (Q&R MEB (DSER 49)

3.7.3.12 Buried Seismic Category I Piping System and Tunnels

Many underground elements like piping, tunnels, reinforced concrete electrical cable ducts, etc., of vital importance need to be designed for accidental conditions such as seismic shock waves passing through the soil medium supporting the element.

The buried piping was designed using the ASME Section III, 1977 through Winter 1978 Addenda, stress equations (refer to Tables 3.2-1 and 3.2-3 for the applicable safety class and ASME code). The following stresses were calculated for buried elements: a) for straight portions of the element, the axial stress; b) at bends, the axial and bending stresses.

For the straight section of elements, Newmark (Reference 12) has presented the following relationship for the maximum axial strain in the element:

a. When particle displacement, P, is along the direction of propagation of wave,

$$\varepsilon_m = \frac{Vm}{c} \qquad (3.7\text{-}12)$$

b. When particle displacement, p, is perpendicular to the direction of propagation of wave,

$$\varepsilon_m = \frac{Vm}{2c} \qquad (3.7\text{-}12)$$

where:

ε_m = maximum axial strain in the homogenous element,

V_m = maximum particle velocity,

C = apparent shear wave velocity in soil.

The stress, σ , is given by

$$\sigma = \varepsilon m^{E} \qquad (3.7.13)$$

Where E is the modulus of elasticity for the element. The values used for the maximum particle velocity and the apparent shear wave velocity, were

V_m = 0.4 ft/sec for OBE

= 1.0 ft/sec for SSE

C = 2,500 ft/sec

The maximum particle velocity was chosen based on the equation given in Reference 15.

$$\frac{ad}{v^2} = 6 \qquad (3.7.15)$$

where a is the acceleration, v the velocity, and d the displacement. Using the Regulatory Guide 1.60 value of 36 in/g for d, this equation gives a velocity of 48 in/sec/g. For 0.25 g, therefore, the maximum particle velocity is 12 in/sec. The value of the apparent shear wave velocity of 2,500 ft/sec was chosen based on the recommended value given in Reference 12; this is a conservative value for the Wisconsinan glacial till and the interglacial zone present at the site.

At bends, the stresses due to the moment, M, induced in the element are included in addition to the axial stress given by Equation 3.7.13. This moment is given by Reference 11.

$$M = \frac{k\Delta}{2\lambda^2} \qquad (3.7.14)$$

where:

$$\Delta = \frac{\varepsilon_m \lambda_m}{s}$$

$$k = k_o b$$

$$\lambda = \sqrt[4]{k/(4EI)}$$

ℓ_m = maximum slippage length

k_o = modulus of subgrade reaction for fill at the bend

b = width of element on elastic foundation

I = moment of inertia of element

The strains in the buried elements were determined using the effects of a shear wave propagating at 45°F to the buried elements. This maximizes the strains (see Reference 12 from Section 3.7).

Piping that enters the building foundation is rigidly connected to the foundation penetration sleeve as shown by Detail of Pipe Attachment (Figure 3.7-99). The pipe is modeled as a beam on an elastic foundation and the pipe stresses are checked for the relative displacements between the supports.

Buried electrical duct runs are rigidly connected to electrical manholes as shown by the typical connection detail (Figures 3.7-100 and 3.7-101). Differential movement between manhole and duct is considered in the concrete duct reinforcing steel design by modeling the duct as a beam on an elastic foundation. (Q&R 220.36)

3.7.3.13 Interaction of Other Piping With Seismic Category I Piping

The seismic-induced effects of non-Seismic Category I piping on Seismic Category I piping are accounted for by including in the analysis of the Seismic Category I piping a length of the non-Category I piping to the first anchor beyond the point where the change in category occurs. A sufficient number of restraints on the non-Seismic Category I piping are seismically designed. At least one restraint in each global direction is required. The axial direction restraint can be located on the Seismic Category I piping adjacent to the pipe category change, since this will also restrain the non-Seismic Category I piping in the axial direction. These criteria meet the requirement of Regulatory Guide 1.29.

3.7.3.14 Seismic Analyses for Reactor Internals

This mathematical modeling of the RPV and internals 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 and the effects of both bending and shear are included.

Mass points are located at all points of critical interest such as anchors, supports, and points of discontinuity, etc. In addition, mass points are chosen such that the total mass of the structure is generally uniformly distributed over all the mass points and the full range of frequency of response of interest is adequately represented. Further, 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 load. 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, in-core guide tubes and housing, 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 details of the hydrodynamic mass derivation are given in Reference 10. 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 vertical analysis is performed. Dynamic loads due to vertical motion are added to or subtracted from the static weight of components, whichever is more conservative. The two rotational coordinates about each node point are excluded because the moment contribution of rotary inertia from surrounding nodes is negligible. Since all deflections are assumed to be within the elastic range, the rigidity of some components may be accounted for by equivalent linear springs.

The seismic analysis of the RPV and internals employs a linear dynamic model consisting of a detailed representation of the RPV and internals combined with an overall model of the RPV support and containment structure. Such a configuration accounts for the dynamic interaction under dynamic loadings such as seismic. The composite dynamic model is referred to as the "primary structure" model.

Sufficient details of the RPV internals components are included in the primary structure dynamic model to enable the generation of representative component interface loads. The interface loads are, in turn, applied to more detailed component stress model in which inelastic behavior is allowed per the ASME III Code. The inelastic response of the components do not alter the linear response of the primary structure.

On the substructure and component levels, the ASME Code allows stresses above the elastic limits depending on the subsystem. Typical examples are the core support plate and top guide which are traditionally represented by single mass points in the RPV portion of the primary system dynamic models. Such inelastic stresses are quite localized and have insignificant effect on the linear response of the primary structure. All local inelastic stresses are verified to be within the ASME Code allowables by appropriate detailed substructure or component analyses. (Q&R MEB (DSER) 51)

The shroud support plate is loaded in its own plane during a seismic event and is hence extremely stiff. It 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.

The damping values are given in Table 3.7-1.

3.7.3.15 Analysis Procedure for Damping

Damping values used for seismic subsystem analysis are in accordance with Subsection 3.7.1.3. Composite damping is not used in the analysis of subsystems.

Alternate damping criteria as specified in ASME Code Case N-411 (Reference 25) can be used for piping subsystems within the limitations documented in References 26 and 27. In particular, the following conditions apply for the use of Code Case N-411 damping:

1) Code Case N-411 damping values are applicable only to building-filtered response spectrum loads (seismic and hydrodynamic); the Code Case damping values shall not be used in time-history analyses such as annulus pressurization, hydraulic transient due to mains steam stop valve closure, etc.

2) When using Code Case N-411 damping values in an analysis, they shall be used in their entirety, and shall not be a mixture of Code Case criteria and Regulatory Guide 1.61 criteria.

3) If, as a result of using Code Case N-411 criteria, pipe supports are moved, modified, or eliminated, any increased piping displacements due to increased system flexibility shall be checked for adequate clearance with adjacent structures, components and equipment.

4) Code Case N-411 damping is limited to frequencies below 33 H_z.

3.7.4 Seismic Instrumentation

3.7.4.1 Comparison with Regulatory Guide 1.12

The following seismic instrumentation program is provided. It is designed in accordance with Regulatory Guide 1.12, "Instrumentation for Earthquakes", for plants with an SSE of less than 0.3 g. The SSE maximum ground acceleration value at the foundation level for Clinton Power Station has been set at 0.25 g.

3.7.4.2 Location and Description of Instrumentation

The instrumentation locations have been chosen to allow meaningful correlation between the recorded accelerations and those calculated using the analytical model of the structure. In addition, the quantities and locations of the instruments are in conformity with Regulatory Guide 1.12.

3.7.4.2.1 Time-History System

The triaxial accelerometers are oriented such that the three axes correspond to the major axes of the analytical model used in the seismic design of the station. Each of five triaxial accelerometers provides input into the Central Recording Unit in the main control room. Four of these triaxial accelerometers (b,c,d,e below) are specifically monitored by the Central Recording Unit in the main control room. At an acceleration of 0.02 g along any of the three axes, the Central Recording Unit starts recording. This acceleration is chosen to screen out minor disturbances while at the same time allowing sufficient sensitivity for appreciable tremors. These accelerometers are located as follows:

a. A triaxial accelerometer is placed approximately 1640 feet north of the plant east-west baseline and 210 feet west of the plant north-south baseline on a small underground concrete pad.

b. A triaxial accelerometer is located at the containment basemat elevation (712 feet) at the azimuth 90°, near the containment wall on the north side of the wall separating the auxiliary building and fuel building. The seismic response at this point is the same as the basemat inside the containment. This sensor also provides data input to the response spectrum analyzer described in Subsection 3.7.4.2.3.

c. A triaxial accelerometer is located on the containment wall inside of containment at elevation 851 feet, azimuth 90°.

d. A triaxial accelerometer is located on the control building floor at approximate elevation 737 feet near column-row AC-128.

e. A triaxial accelerometer is located in containment on the drywell wall, elevation 779' 10", azimuth 90°. Data from this sensor is automatically recorded by the Seismic Central Recorder and may be viewed by the Seismic Data Analyzer described in Section 3.7.4.2.3.

When the Seismic Central Recorder is started for an event, an annunciator in the main control room and an indicating light on the Seismic Warning Panel are actuated. Detailed time-history analysis of the seismic event can be obtained using the Seismic Data Analyzer and/or Seismic Data Printer. The seismic information can be viewed on the Seismic Data Analyzer (PC screen) and/or printed.

3.7.4.2.2 Seismic Switch

A triaxial seismic switch is placed at the same location as the triaxial accelerometer located at the containment basemat (elevation 712 feet, azimuth 90°) just outside the containmentwall (in the auxiliary building). The central control unit actuates an indicating light and an annunciator in the main control room if the zero period acceleration (of the OBE response spectrum at that location) is exceeded in any of the three axes. The axes are oriented identically with those of the time-history accelerograph sensors.

3.7.4.2.3 Response Spectrum Analysis

A passive response spectrum recorder is located at the circulating water screen house (Seismic Category I Structure independent of the power plant structure). This instrument consists of 12 reeds tuned to different frequencies encompassing the significant portion of the seismic design spectrum for the plant. Peak responses are recorded on scratch plates.

A Seismic Central Recorder, Seismic Data Analyzer, and Seismic Data Printer capable of computing and plotting desired response spectra is provided in the Main Control Room. When an event of sufficient magnitude occurs, the Seismic Central Recorder is automatically activated. Data from all five accelerometers identified in Section 3.7.4.2.1 are automatically recorded and once down loaded, may be viewed with the Seismic Data Analyzer.

The Seismic Data Analyzer compares the sensor data against the analytical response spectra for the sensor location which is permanently stored in the Data Analyzer. If any axis of the sensors exceeds these stored values, an indicating light and an annunciator in the main control room are actuated. One annunciator will inform the operator if any OBE value is exceeded, and another annunciator if any SSE value is exceeded. The Seismic Data Analyzer provides detailed information as to the frequencies, axis, and sensor which resulted in actuation of the annunciator.

Time-history data from any of the five sensors identified in Section 3.7.4.2.1 can be manually downloaded form the Seismic Central recorder to the Seismic Data Analyzer for viewing or printing.

3.7.4.2.4 Peak Accelerographs

Three triaxial peak accelerographs, each of which measures the absolute peak acceleration in three orthogonal directions coinciding with the principal axes of the analytical model, are placed at the following locations:

a. Standby Liquid Control Tank (Seismic Category 1 Reactor Equipment),

b. Seismic Category 1 piping connected to an RHR heat exchanger, and

c. Diesel generator oil storage tank (Seismic Category I equipment outside of containment).

3.7.4.2.5 Instrument Performance

All instruments are designed to perform their functions satisfactorily over the expected range of environmental conditions, including temperature, humidity, pressure and radiation.

A panel in the main control room contains the Seismic Central Recorder, indicating lights, Seismic Data Analyzer, and a power supply. The cabinet and this equipment are qualified in accordance with the requirements of IEEE 344.

Battery backup power is automatically provided to operate the equipment on loss of the normal power source with the exception of the printer. The printer is not required in the event of power loss because the seismic information is also available from the Seismic Data Analyzer (PC)

Display. An annunciator in the main control room is actuated on loss of the normal power source.

Built-in test equipment is provided to allow complete in-place testing Central Recorder, Seismic Data Analyzer, and the five triaxial accelerometers.

3.7.4.3 Control Room Operator Notification

As described previously, annunciators will alert the operator when

a. A control unit starts the Central Recorder

b. The seismic switch senses that the OBE acceleration has been exceeded.

c. After data is downloaded to the Seismic Data Analyzer and has detected that an OBE acceleration for the on-line accelerometers described in Subsection 3.7.4.2.3 has been exceeded.

d. After data is downloaded to the Seismic Data Analyzer and has detected that an SSE acceleration for these sensors has been exceeded.

e. Loss of the normal power source occurs.

The operator can download the data from the Central Recorder to the Seismic Data Analyzer and examine the Seismic Data Analyzer Display and/or the printer output of the time history records in the control room. In addition, the data from the passive response spectrum analyzer in the circulating water screen house and the three peak accelerographs may be examined.

If the OBE maximum acceleration has been exceeded, the operator initiates shutdown of the station. Besides the annunciators, the recorded data is available for analysis of the seismic event.

3.7.4.4 Comparison of Measured and Predicted Responses

The measured response spectra will be compared with the corresponding predicted response spectra for the locations noted in Subsection 3.7.4.2. Agreement between the measured response spectra and the predicted response spectra, or measured response spectra being smaller than the predicted response spectra, would authenticate the capability of the plant to continue operation without undue risk to the health and safety of the public.

In the event that the measured response spectra greatly exceed the predicted response spectra, additional evaluation will be performed. This evaluation could include detailed analyses using recorded time histories, remodeling and physical inspection.

3.7.5 References

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TABLE 3.7-1
DAMPING VALUES

ITEM EQUIPMENT, OR STRUCTURE	DAMPING, PERCENT OF CRITICAL (1) OBE	SSE
BALANCE OF PLANT		
Equipment and large diameter piping systems, pipe diameter greater than 12 inches	2	3
Small diameter piping systems, diameter less than or equal to 12 inches	1	2
Welded steel structures	2	4
Bolted steel structures	4	7
Prestressed concrete structures	2	5
Reinforced concrete structures	4	7
Soil	(2)	(2)
NSSS		
Welded Structural Assemblies (Equipment and Supports)	2	3
Vital Piping Systems		
- Diameter Greater Than 12 in.	2	3
- Diameter Less Than or Equal to 12 in.	1	2
Reactor Pressure Vessel, Support Skirt, Shroud Head, Separator and Guide Tubes	2	4
Fuel	6	6

(1) Alternate critical damping values for piping systems may be used as described in Section 3.7.1.3.2.

(2) Since strain-dependent soil properties are used for the soil-structure interaction, no specific damping values are included.

TABLE 3.7-2
CONTAINMENT MODEL FOR HORIZONTAL EXCITATION - MODAL FREQUENCIES AND PARTICIPATION FACTORS

MODE	FREQUENCY (hertz)	PARTICIPATION FACTORS	
		E-W EXCITATION	N-S EXCITATION
1	4.86	26.93	0.00
2	4.87	0.00	-26.91
3	4.98	40.98	-1.55
4	4.98	1.81	35.04
5	5.10	-4.75	-0.01
6	7.68	5.88	-0.00
7	7.80	-0.00	-6.21
8	8.28	-2.77	-0.00
9	8.35	0.00	-2.56
10	11.02	2.67	-0.02
11	11.23	-0.83	15.23
12	13.46	15.24	0.00
13	14.77	0.99	-14.89
14	16.96	21.51	-0.04
15	21.60	0.72	0.61
16	21.79	0.00	0.60
17	24.41	-6.88	0.00
18	24.41	0.00	-6.88
19	26.77	-5.98	0.00
20	26.98	0.00	6.02
21	27.55	0.00	-4.10
22	27.99	5.56	0.00

TABLE 3.7-3
ONE-UNIT MAIN STRUCTURE MODEL FOR HORIZONTAL EXCITATION – MODAL FREQUENCIES AND PARTICIPATION FACTORS

MODE	FREQUENCY (hertz)	PARTICIPATION FACTORS E-W EXCITATION	PARTICIPATION FACTORS N-S EXCITATION
1	1.95	-9.00	0.03
2	2.47	-0.06	-9.16
3	4.97	0.85	-0.20
4	6.80	-103.00	8.50
5	7.58	-15.74	-97.69
6	8.75	-24.06	28.23
7	12.82	-9.25	41.82
8	14.72	-37.76	6.50
9	17.57	28.40	12.21
10	18.16	-20.15	0.92
11	20.52	4.10	-46.64
12	22.43	-28.08	0.59
13	23.84	-2.02	3.66
14	27.36	-6.00	-15.65
15	29.42	9.01	8.83
16	30.13	6.32	-19.69
17	33.71	1.02	1.24

TABLE 3.7-4
TWO-UNIT MAIN STRUCTURE MODEL FOR HORIZONTAL EXCITATION - MODAL FREQUENCIES AND PARTICIPATION FACTORS

MODE	FREQUENCY (hertz)	PARTICIPATION FACTORS	
		E-W EXCITATION	N-S EXCITATION
1	1.96	-12.77	.01
2	3.50	-0.21	14.99
3	3.51	0.57	1.52
4	5.49	2.26	-0.04
5	6.60	133.07	-2.48
6	7.62	4.53	133.40
7	9.40	27.83	-9.07
8	12.27	0.30	-40.65
9	13.45	34.63	-0.07
10	15.50	-52.73	0.58
11	17.65	22.52	6.09
12	19.80	3.92	-60.99
13	21.79	28.87	8.28
14	22.71	-24.08	3.59
15	26.89	-1.59	23.14
16	27.97	-2.89	22.40
17	29.99	9.85	8.20
18	31.07	3.04	-2.62

TABLE 3.7-5
CONTAINMENT MODEL FOR VERTICAL EXCITATION - MODAL FREQUENCIES AND PARTICIPATION FACTORS

MODE	FREQUENCIES (hertz)	PARTICIPATION FACTOR
1	2.00	2.27
2	5.00	-2.58
3	8.00	-3.44
4	10.93	6.08
5	10.94	-6.48
6	12.90	-10.34
7	12.96	2.31
8	12.99	-1.61
9	13.02	2.63
10	13.35	8.81
11	13.54	-21.74
12	13.93	-7.74
13	13.98	2.21
14	14.05	-1.98
15	14.62	-23.10
16	16.20	-26.47
17	16.98	-1.20
18	17.35	14.23
19	18.12	4.48
20	19.80	-2.78
21	19.91	2.59
22	19.97	1.05
23	20.46	-7.16
24	22.09	-2.36
25	24.23	-3.99
26	26.08	-1.75
27	29.25	18.37
28	32.15	-2.51

TABLE 3.7-6
ONE-UNIT MAIN STRUCTURE MODEL FOR VERTICAL EXCITATION - MODAL FREQUENCIES AND PARTICIPATION FACTORS

MODE	FREQUENCIES (hertz)	PARTICIPATION FACTOR
1	2.00	13.38
2	2.00	7.85
3	3.00	10.19
4	4.00	-13.60
5	4.00	-8.14
6	5.99	14.07
7	6.00	-1.54
8	7.00	-14.30
9	8.97	-16.58
10	8.98	21.48
11	8.99	1.64
12	9.00	5.46
13	11.97	13.77
14	11.98	17.48
15	12.91	-20.80
16	12.93	-22.35
17	13.90	14.03
18	14.67	-34.52
19	14.85	34.37
20	14.98	5.78
21	16.90	33.29
22	17.59	47.56
23	17.7	42.74
24	18.38	-33.69
25	19.00	-61.04
26	19.23	32.77
27	19.96	-6.19
28	20.61	39.36
29	21.28	-45.43
30	22.74	-8.19
31	23.15	10.23
32	24.12	38.00
33	24.92	-7.59
34	26.51	17.71
35	27.20	11.47
36	27.83	-6.42
37	28.85	32.61
38	29.76	-14.67
39	32.55	27.63

TABLE 3.7-7
TWO-UNIT MAIN STRUCTURE MODEL FOR VERTICAL
EXCITATION - MODAL FREQUENCIES AND PARTICIPATION FACTORS

MODE	FREQUENCY (hertz)	PARTICIPATION FACTOR
1	2.00	-9.80
2	2.00	-17.45
3	3.00	13.72
4	4.00	-10.19
5	4.00	17.75
6	5.99	18.63
7	6.00	-22.39
8	6.99	-18.72
9	8.97	-28.17
10	11.96	-18.44
11	12.88	-23.40
12	11.98	29.73
13	13.91	30.63
14	12.92	-15.50
15	14.72	42.69
16	14.82	48.57
17	14.98	7.81
18	16.85	-56.08
19	17.10	71.95
20	17.42	67.11
21	18.55	28.77
22	18.70	69.12
23	19.20	39.32
24	19.96	8.47
25	20.39	32.67
26	21.19	-53.70
27	22.69	-9.60
28	23.08	-6.73
29	24.04	44.60
30	24.89	-12.49
31	26.35	24.21
32	27.15	-11.92
33	27.68	-14.23
34	28.76	38.28
35	29.57	-27.23
36	30.47	19.33
37	32.22	41.22

TABLE 3.7-8
SSE FORCES AND MOMENTS FOR SINGLE-UNIT CONTAINMENT MODEL

	RESPONSE SPECTRUM METHOD		TIME HISTORY METHOD	
MEMBER #	SHEAR (Kips)	MOMENT (Kip-Ft)	SHEAR (Kips)	MOMENT (Kip-Ft)
88	8.4×10^3	1.2×10^6	9.8×10^3	1.2×10^6
89	8.1×10^3	1.0×10^6	9.1×10^3	9.9×10^5
90	7.8×10^3	8.6×10^5	8.4×10^3	8.0×10^5
91	7.2×10^3	7.1×10^5	7.5×10^3	6.3×10^5
92	6.9×10^3	5.6×10^5	6.4×10^3	4.8×10^5
93	5.7×10^3	4.2×10^5	5.3×10^3	3.5×10^5
94	4.6×10^3	3.0×10^5	4.2×10^3	2.3×10^5
95	3.7×10^3	1.9×10^5	3.3×10^3	1.4×10^5
96	2.6×10^3	1.4×10^5	2.3×10^3	1.1×10^5
102	7.7×10^2	5.6×10^4	6.6×10^2	3.2×10^4

(Refer to Figure 3.7-30 for member location)

TABLE 3.7-9

PARAMETERS FOR ANALYSIS OF ROCK-SOIL-STRUCTURE-INTERACTION

(FINITE ELEMENT MODEL)

	COHESIONLESS SOIL	COHESIVE SOILS					COHESIONLESS SOIL	
	COMPACTED STRUCTURAL FILL	RECOMPACTED WISCONSINAN GLACIAL TILL OF WEDRON FORMATION TYPE A MATERIAL (AS COMPACTED)	RECOMPACTED WISCONSINAN GLACIAL TILL OF WEDRON FORMATION TYPE A MATERIAL (SATURATED)	LOESS	WISCONSINAN GLACIAL TILL OF WEDRON FORMATION	INTER-GLACIAL DEPOSITS	SALT CREEK ALLUVIUM	INTERGLACIAL SAND DEPOSITS
DENSITY (pcf)								
Dry density	123	127	128	101	118	115	100	108
Wet density	132	141	144	120	137	131	125	120
POISSON'S RATION								
Dynamic	0.40	0.40	0.40	0.40	0.40	0.40	0.40	0.40
Static	0.30	0.40	0.40	0.40	0.40	0.40	0.40	0.40
STATIC MODULUS OF ELASTICITY (Es)								
In situ modulus (psf)	-	8.0×10^5	2.0×10^5	2.0×10^5	13.1×10^5	15.1×10^5	-	-
Increase with surcharge								
$\frac{dE_s}{d\sigma'_m}$ (psf/psf)	350	0	0	0	0	0	150	260
DYNAMIC MODULUS OF ELASTICITY (psf)								
Single amplitude								
Shear strain = 1.0%	$5{,}600\ (\sigma'_m)^{1/2}$	11×10^5	3×10^5	3×10^5	12×10^5	8×10^5	$2{,}800\ (\sigma'_m)^{1/2}$	$4{,}200\ (\sigma'_m)^{1/2}$
= 0.1%	$36{,}400\ (\sigma'_m)^{1/2}$	39×10^5	8×10^5	8×10^5	34×10^5	31×10^5	$11{,}000\ (\sigma'_m)^{1/2}$	$17{,}000\ (\sigma'_m)^{1/2}$
= 0.01%	$95{,}000\ (\sigma'_m)^{1/2}$	98×10^5	34×10^5	34×10^5	101×10^5	84×10^5	$45{,}000\ (\sigma'_m)^{1/2}$	$62{,}000\ (\sigma'_m)^{1/2}$
= 0.001%	$117{,}600\ (\sigma'_m)^{1/2}$	148×10^5	76×10^5	76×10^5	232×10^5	185×10^5	$53{,}000\ (\sigma'_m)^{1/2}$	$81{,}000\ (\sigma'_m)^{1/2}$
= 0.0001%	$126{,}000\ (\sigma'_m)^{1/2}$	162×10^5	95×10^5	95×10^5	336×10^5	280×10^5	$56{,}000\ (\sigma'_m)^{1/2}$	$84{,}000\ (\sigma'_m)^{1/2}$
STATIC MODULUS OF RIGIDITY (Gs)								
In situ modlus (psf)	-	3.0×10^5	0.7×10^5	0.7×10^5	4.7×10^5	5.4×10^5	-	-
Increase with surcharge								
$\frac{dG_s}{d\sigma'_m}$ (psf/psf)	135	0	0	0	0	0	54	93
DYNAMIC MODULUS OF RIGIDITY (psf)								
Single amplitude								
Shear strain = 1.0%	$2{,}000\ (\sigma'_m)^{1/2}$	4×10^5	1×10^5	1×10^5	4×10^5	3×10^5	$1{,}000\ (\sigma'_m)^{1/2}$	$1{,}500\ (\sigma'_m)^{1/2}$
= 0.1%	$13{,}000\ (\sigma'_m)^{1/2}$	14×10^5	3×10^5	3×10^5	12×10^5	11×10^5	$4{,}000\ (\sigma'_m)^{1/2}$	$6{,}000\ (\sigma'_m)^{1/2}$
= 0.01%	$34{,}000\ (\sigma'_m)^{1/2}$	35×10^5	12×10^5	12×10^5	36×10^5	30×10^5	$16{,}000\ (\sigma'_m)^{1/2}$	$22{,}000\ (\sigma'_m)^{1/2}$
= 0.001%	$42{,}000\ (\sigma'_m)^{1/2}$	53×10^5	27×10^5	27×10^5	83×10^5	66×10^5	$19{,}000\ (\sigma'_m)^{1/2}$	$29{,}000\ (\sigma'_m)^{1/2}$
= 0.0001%	$45{,}000\ (\sigma'_m)^{1/2}$	58×10^5	34×10^5	34×10^5	120×10^5	100×10^5	$20{,}000\ (\sigma'_m)^{1/2}$	$30{,}000\ (\sigma'_m)^{1/2}$

σ'_m -- mean effective stress (psf).

TABLE 3.7-9 (CONT'D)

	COHESIONLESS SOIL	COHESIVE SOILS					COHESIONLESS SOIL	
	COMPACTED STRUCTURAL FILL	RECOMPACTED WISCONSINAN GLACIAL TILL OF WEDRON FORMATION TYPE A MATERIAL (AS COMPACTED)	RECOMPACTED WISCONSINAN GLACIAL TILL OF WEDRON FORMATION TYPE A MATERIAL (SATURATED)	LOESS	WISCONSINAN GLACIAL TILL OF WEDRON FORMATION	INTER-GLACIAL DEPOSITS	SALT CREEK ALLUVIUM	INTERGLACIAL SAND DEPOSITS
DAMPING								
Percent of critical damping single amplitude								
Shear strain = 1.0%	16	20	20	20	20	20	21	28
= 0.1%	14	9	15	15	9	9	10	13
= 0.01%	6	5	10	10	5	5	3	4
= 0.001%	2	3	6	6	3	3	1	1.5
= 0.0001%	1	2.5	4	4	2.5	2.5	0.5	0.5

NOTES:

1. The static modulus of elasticity values for cohesive soils were calculated based on the constrained modulus derived from the reloading portion of the consolidation curve

2. Pre-Illinoian cohesive deposits include glacial and lacustrine deposits.

3. Pre-Illinoian cohesionless deposits include Mahomet Valley deposits.

4. The selected parameters reflect both the results of geophysical and laboratory tests performed during this investigation and results published and previously developed for similar soils.

TABLE 3.7-9 (CONT'D)

	COHESIVE SOIL			COHESIONLESS SOIL	
	ILLINOIAN GLACIAL TILL	LACUSTRINE DEPOSITS	PRE-ILLINOIAN DEPOSITS	PRE-ILLINOIAN DEPOSITS	ROCK*
DENSITY (pcf)					
Dry density	138	123	130	107	156
Wet density	150	134	145	126	159
POISSON'S RATIO					
Dynamic	0.35	0.35	0.35	0.4	0.29
Static	0.35	0.35	0.35	0.4	0.29
STATIC MODULUS OF ELASTICITY (Es)					
In situ modulus (psf)	43.6×10^5	24.9×10^5	42.4×10^5	110×10^5	0.7 to 3.8×10^8
Increase with surcharge					
$\frac{dE_s}{d\sigma'_m}$ (psf/psf)	0	0	0	1100	0
DYNAMIC MODULUS OF ELASTICITY (psf)					
Single amplitude					3.6 to 7.8×10^8
Shear strain = 1.0%	22×10^5	22×10^5	22×10^5	$28{,}000\ (\sigma'_m)^{1/2}$	0
= 0.1%	81×10^5	76×10^5	70×10^5	$95{,}000\ (\sigma'_m)^{1/2}$	
= 0.01%	270×10^5	192×10^5	208×10^5	$174{,}000\ (\sigma'_m)^{1/2}$	
= 0.001%	702×10^5	392×10^5	540×10^5	$218{,}000\ (\sigma'_m)^{1/2}$	
= 0.0001%	1620×10^5	648×10^5	923×10^5	$238{,}000\ (\sigma'_m)^{1/2}$	
STATIC MODULUS OF RIGIDITY (Gs)					
In situ modlus (psf)	16.1×10^5	9.2×10^5	15.7×10^5	40×10^5	0.3 to 1.5×10^8
Increase with surcharge					
$\frac{dG_s}{d\sigma'_m}$ (psf/psf)	0	0	0	392	0
DYNAMIC MODULUS OF RIGIDITY (psf)					
Single amplitude					1.4 to 3.0×10^8
Shear strain = 1.0%	8×10^5	8×10^5	8×10^5	$10{,}000\ (\sigma'_m)^{1/2}$	0
= 0.1%	30×10^5	28×10^5	26×10^5	$34{,}000\ (\sigma'_m)^{1/2}$	
= 0.01%	100×10^5	71×10^5	77×10^5	$62{,}000\ (\sigma'_m)^{1/2}$	
= 0.001%	260×10^5	145×10^5	200×10^5	$78{,}000\ (\sigma'_m)^{1/2}$	
= 0.0001%	600×10^5	240×10^5	342×10^5	$85{,}000\ (\sigma'_m)^{1/2}$	
DAMPING					
Percent of critical damping single amplitude					
Shear strain = 1.0%	22	20	20	20	1 to 2
= 0.1%	16	9	12	10	
= 0.01%	7.5	4.5	7.5	3	
= 0.001%	4	3	4	2	
= 0.0001%	3	2.5	3	1	

TABLE 3.7-9 (CONT'D)

* These values are valid for strain levels on the order of 10^{-4} to 10^{-5} percent.

TABLE 3.7-10
(This Table has been Deleted.)

TABLE 3.7-11

NUMBER OF DYNAMIC RESPONSE CYCLES EXPECTED DURING A SEISMIC EVENT

Frequency Band Hz	0+-10	10-20	20-50
Total Number of Seismic Cycles	168	359	643
No. of Seismic Cycles 0.5% cycles between 75% and 100% of Peak Loads	0.8	1.8	3.2
No. of Seismic Cycles 4.5% cycles between 50% and 75% of Peak Loads	7.5	16.2	28.9

TABLE 3.7-12
CIRCULATING WATER SCREEN HOUSE MODEL FOR HORIZONTAL EXCITATION - MODAL FREQUENCIES AND PARTICIPATION FACTORS

MODE	FREQUENCY (Hertz)	PARTICIPATION N-S EXCITATION	FACTORS E-W EXCITATION
1	10.94	48.50	0.02
2	17.88	-0.03	50.53
3	20.41	8.85	-0.03
4	24.24	-11.13	-0.11
5	31.88	0.43	-7.90
6	36.91	4.38	0.68
7	38.72	-3.94	0.19
8	40.36	1.14	1.45
9	56.19	-5.24	0.14

TABLE 3.7-13
CIRCULATING WATER SCREEN HOUSE MODEL FOR VERTICAL EXCITATION - MODAL FREQUENCIES AND PARTICIPATION FACTOR

MODE	FREQUENCIES (Hertz)	PARTICIPATION FACTOR
1	3.012	-14.44
2	3.014	0.00
3	3.641	174.00
4	5.023	0.00
5	5.026	6.15
6	6.934	0.00
7	6.956	7.22
8	6.988	0.00
9	6.991	4.23
10	7.031	0.00
11	7.032	-3.31
12	9.891	-7.89
13	9.892	-0.01
14	9.965	-2.41
15	9.969	0.00
16	10.036	1.99
17	10.039	0.00
18	13.363	51.50
19	15.298	0.00
20	15.929	3.90
21	18.634	-8.46
22	18.684	0.00
23	19.843	-3.51
24	19.883	0.00
25	21.843	22.84

TABLE 3.7-14
(Q&R 220.21)
STRAIN DEPENDENT SOIL PROPERTIES

	OBE - HORIZ.			SSE - HORIZ.		
Layer	Strain (%)	Damp.	$Gx10^3$ (ksf)	Strain (%)	Damp.	$Gx10^3$ (ksf)
1	.00031	.025	10.25	.00076	.028	8.78
2	.00120	.031	7.89	.00370	.037	5.39
3	.00266	.034	6.08	.00895	.044	3.79
4	.00469	.039	4.93	.01607	.054	3.00
5	.01713	.043	4.17	.02539	.062	2.41
6	.00995	.045	3.61	.03599	.069	2.02
7	.01629	.054	2.51	.05715	.077	1.42
8	.01898	.057	2.36	.06389	.079	1.35
9	.02132	.059	2.24	.06768	.079	1.32
10	.02313	.060	2.16	.07575	.082	1.26
11	.00315	.056	16.71	.01056	.076	9.79
12	.00371	.059	15.75	.01217	.079	9.25
13	.00432	.062	14.86	.01388	.082	8.75
14	.00497	.065	14.04	.01569	.085	8.30
15	.00568	.067	13.24	.01758	.088	7.84
16	.00644	.069	12.50	.01957	.090	7.43
17	.00723	.070	11.82	.02161	.093	7.05
18	.00805	.072	11.22	.02370	.095	6.70
19	.00890	.073	10.65	.02581	.097	6.38
20	.00977	.075	10.13	.02789	.098	6.08
21	.01047	.076	9.82	.02990	.100	5.81
22	.01111	.077	9.60	.03131	.101	5.69
23	.01172	.079	9.39	.03260	.102	5.59
24	.01230	.080	9.21	.03384	.104	5.49
25	.01989	.088	6.01	.05419	.107	3.58
27	.02076	.088	5.91	.05793	.109	3.48
28	.02122	.089	5.85	.06124	.111	3.40
29	.02143	.089	5.83	.06394	.112	3.33
30	.02140	.089	5.83	.06607	.113	3.29
31	Bedrock					

TABLE 3.7-15
(Q&R 220.28)
PERIODS OF THE RESPONSE SPECTRA

NUMBER	PERIOD (seconds)
1	0.020
2	0.030
3	0.040
4	0.045
5	0.050
6	0.055
7	0.060
8	0.065
9	0.070
10	0.075
11	0.080
12	0.085
13	0.090
14	0.095
15	0.100
16	0.110
17	0.120
18	0.130
19	0.140
20	0.150
21	0.160
22	0.170
23	0.180
24	0.190
25	0.200
26	0.220
27	0.240
28	0.260
29	0.280
30	0.300
31	0.320
32	0.340
33	0.360
34	0.380
35	0.400
36	0.450
37	0.500
38	0.550
39	0.600
40	0.700
41	0.800
42	0.900
43	1.000
44	1.100
45	1.200
46	1.300
47	1.400
48	1.500
49	1.700
50	2.000

TABLE 3.7-16
(Q&R 220.31)
COMPARISON OF TYPICAL SHEAR WALL DESIGN BASIS FORCES
TO THOSE INDUCED BY THE 5% ACCIDENTAL TORSION FOR SSE

SPRING NO.	DESIGN LOAD	ACCIDENTAL TORSION LOAD	% INCREASE
X-Direction Shear Walls			
1027	15448.0	1974.0	12.8
102023	14692.0	1589.0	10.8
1019	13426.0	1206.0	9.0
204023	12965.0	1146.0	8.8
204005	11201.0	1.4	0.0
1009	10005.0	38.8	0.4
405019	8577.0	751.0	8.8
708001	7516.0	303.0	4.0
307001	6645.0	929.0	14.0
203031	5023.0	342.0	6.8
204015	4608.0	275.0	6.0
507007	3138.0	9.0	0.3
304005	2295.0	248.0	10.8
507021	993.0	46.0	4.6
Y-Direction Shear Walls			
304004	15685.0	82.0	0.5
1018	14298.0	1757.0	12.3
203006	13416.0	1186.0	8.8
102002	12484.0	197.0	1.6
2012	10906.0	314.0	2.9
708002	9940.0	376.0	3.8
405002	8459.0	636.0	7.5
405026	7213.0	619.0	8.6
204016	5358.0	533.0	9.9
102018	4308.0	564.0	13.1
203026	2891.0	121.0	4.2
2042	1591.0	153.0	9.6
1060	800.0	85.0	10.6

ATTACHMENT A3.7 (Q&R 220.15)

Dr. V. P. Drnevich Letter of
February 23, 1982 on
GRANULAR STRUCTURAL FILL

(Text of letter Dr. Vincent P. Drnevich to Dr. Terje Preber on February 23, 1982, on Granular structural fill, Illinois Power Company, Clinton Station, Job No. 5646-017-07).

I have received the background information on the granular structual fill which you sent on February 10, 1982. The information was from the Clinton Power Station-Final Safety Analysis Report, Amendment 3, April 1981. The information included: results of in-place density test measurements, particle size distribution curves, and information on mineral content. From these data, I was able to classify the structrual fill material and to establish the parameters from which to estimate initial tangent shear moduli.

I have performed resonant column tests in the past on material similar in nature to this structural fill. In addition, a colleague at the University of Kentucky, Dr. Bobby O. Hardin, completed a fairly detailed study on the Shear Modulus of Gravels for the U.S. Air Force. The final report on Contract F29601-73-0-0064, September 1973, was used to support my calculations to estimate the initial tangent shear modulus for this structural fill.

The process of estimating the initial tangent shear modulus is necessary before one can independently establish valves of K2. Shear moduli were calculated by use of two independent empirical equations; one from the State-of-the-Art paper by Hardin in the ASCE Specialty Conference on Earthquake Engineering and Soil Dynamics in Pasadena in 1978 and the other from the above reference report on gravels. In both methods, the significant parameters were varied to ascertain the sensitivity of shear modulus (and K_2) to the parameters. From these calculations, it is quite evident that a K_2-value of 100 is a very reasonable and realisitic value to use for design purposes. I would expect that if very accurate insitu seismic tests were to be performed on this structural fill, that the values of K_2 back calculated from the measured shear wave propagation velocities would be approximately 100.

I am pleased to be of assistance to you on this matter. If clarification on any of the above items is needed, I would be happy to provide it.

(The four graphs submitted with the Q&R will not be included in the USAR).

1)TABLE A3.7-1
EVALUATION OF GEOPHYSICAL DATA

SITE'	SOIL CONDITIONS	BLOW COUNT	DEPTH RANGE (FT)	COMPRESSIONAL WAVE VELOCITY RANGE (fps)	SHEAR WAVE VELOCITY RANGE (fps)	POISSONS RATIO	
						RANGE	AVERAGE
Fermi III (SDIL)	Till	30-80	0-30	5500-6900	900-1800	0.485-0.463	0.47
Sterling (WPHOLE)	Till	100/6"-refusal	0-80	7400-7900	2150-2700	0.434-.0454	0.44
Attica (SDIL)	Clay	4-8	10-40	5200-6400	800-900	.488-0.490	0.49
	Till (ML)	50-100	40-50	6600	1100	0.485	0.48
	Sands (SP-GM)	20	50-65	7200	2100	0.454	0.45
	Till (ML-SM)	50-150	55-90	8000	1300	0.486	0.49
Bailly (SDIL)	G1 Lac Clay	5-30	20-100	5600	900	0.48	0.48
	G1 Lac Clay	30-200	20-120	5800-6200	1150-1400	0.47-0.48	0.48
	G1 Lac Clay	50-200	120-140	5800	1150	0.48	0.48
	Glacial Till	50-150	120-160	6200	1600	0.46	0.46
LaSalle (SDIL)	Glacial Till (Wisconsinan)	10-100	40-115	5500-6400	950-1450	0.45-0.485	0.475
Clinton	Glacial Till (Illinoian)	12-200/6"	50-170	7500	2000-2100[1]	0.46[2]	

1) Calculated

2) Estimated

3.8 DESIGN OF SEISMIC CATEGORY I STRUCTURES

3.8.1 Concrete Containment

3.8.1.1 Description of the Containment

3.8.1.1.1 General

The basic objective of the containment system is to provide the capability, in the unlikely event of the postulated design-basis loss-of-coolant accident (LOCA), to limit the release of fission products to the station site environs so that offsite doses are in compliance with the values specified in 10 CFR 50.67. In addition to the containment, a standby gas treatment system (SGTS) is installed to process any leakage from the containment via a filter or purge system, automatically or manually.

To meet the basic safety objective, several subsidiary objectives are achieved by the system or one or more of its components, including the following:

a. The containment system has the capability of withstanding the conditions which could result from any of the postulated design-basis accidents for which the containment system is assumed to be functional, including the largest amount of energy release and mass flow associated with the accident.

b. The containment system is capable of withstanding the effects of the metal-to-water reaction and other chemical reactions subsequent to any postulated design-basis accident for which the containment system is assumed to be functional, consistent with the performance objectives of the nuclear safety systems and engineered safety features.

c. The containment system has the capability to maintain its functional integrity during any postulated design event, including protection against missiles from internal or external sources, excessive motion of pipes, jet forces associated with the flow from the postulated rupture of any pipe within the containment, and actuations of safety/relief valves.

d. The containment system has the capability to be filled with water to a level above the active core as an accident recovery method for any postulated design-basis accident in which a breach of the nuclear system primary barrier cannot be sealed.

e. The containment system, in conjunction with other nuclear safety systems and engineered safety features, has the capability to limit leakage during any of the postulated design-basis accidents for which it is assumed to be functional, such that offsite doses do not exceed the guideline values.

f. In the containment, the suppression pool has the means to rapidly condense the steam flow resulting from the design-basis accident which is the rupture of a main steamline inside the drywell.

g. The containment system has the means to conduct the flow from postulated pipe ruptures to the suppression pool, to distribute such flow uniformly throughout the pool, and to limit pressure differentials between the drywell and the containment during the various post-accident cooling modes.

h. Rapid closing, redundant isolation valves will maintain containment leakage at or below permissible limits for all postulated conditions by providing an effective barrier in pipes and ducts that penetrate the containment.

i. The containment has the capability to be periodically leak tested as may be appropriate to confirm the integrity of the containment at the peak transient pressure resulting from the postulated design-basis accident.

j. The containment system has the capability to store sufficient water to supply the requirements of the core standby cooling system.

The suppression pool serves as the principal source of coolant for the low-pressure core spray (LPCS) and the low-pressure core injection (LPCI) mode of the residual heat removal (RHR) system.

3.8.1.1.2 Containment Structure

The containment, shown in Figures 3.8-1 and 3.8-2, consists of a right circular cylinder with a hemispherical domed roof and a flat base slab. It is constructed of reinforced concrete and completely lined on the inside of the walls and dome with 1/4-inch stainless steel plate below elevation 735 feet 0 inch and with carbon steel plate of at least 1/4 inch thickness above elevation 735 feet 0 inch.

The principal dimensions of the containment are:

a. height above basemat: 215 feet 0 inch;

b. inside diameter: 124 feet 0 inch;

c. wall thickness: 3 feet 0 inch;

d. dome thickness: 2 feet 6 inches; and

e. mat thickness: 9 feet 8 inches.

The containment structure supports the polar crane, galleries, and the access ramp to the refueling floor. The lower section of the containment acts as the outer boundary of the suppression pool. Two double-door personnel locks, one located at the refueling floor and the other located at the grade floor, permit access to the containment. An equipment hatch is located at the grade floor. The equipment hatch is sealed during normal operation, or at other times when primary containment is required.

The containment wall is reinforced in the hoop, diagonal and meridional directions as shown in Figure 3.8-3. Wall reinforcement is deflected around small penetration sleeves to account for localized stress concentrations. The wall around the equipment hatch and personnel locks is thickened to 6 feet 0 inch, and additional reinforcement is provided. Reinforcing details around these penetrations are shown in Figure 3.8-4. Tangential and transverse shear reinforcement are provided where necessary.

The dome is reinforced in two directions as shown in Figure 3.8-3. Orthogonal grid type reinforcement is provided within a radius of 45 feet from the apex of the dome. The remaining portion of the dome is reinforced in the hoop and meridional directions.

The containment base slab is continuous with the adjacent auxiliary and fuel building base slabs and is reinforced at top and bottom with reinforcing steel as shown in Figures 3.8-5 through 3.8-9.

3.8.1.1.3 Containment Penetrations

To maintain the containment pressure boundary, containment penetrations have the following design characteristics:

a. capability to withstand peak transient temperatures;

b. capability to withstand the forces caused by the impingement of the fluid from the break of the largest local pipe or connection without failure;

c. capability to accommodate the thermal and mechanical stresses which may be encountered during all modes of operation without failure;

d. capability to withstand the design pressure; and

e. capability to act as a pipe support.

The approximate number and sizes of the principal containment penetrations are shown in Table 3.8-5. Locations of these penetrations are shown in Figure 3.8-10.

3.8.1.1.3.1 Pipe Penetrations

Pipe penetrations for process pipes which pass through the containment and drywell walls may be classified into three types. Type 1 is used for high-energy lines requiring guard pipes when passing through both the containment and drywell walls. Types 2 and 3 are used for the remainder of process pipes which pass through the containment. Figure 3.8-11 shows the basic design of the three penetration types along with the inclined fuel transfer tube detail.

Type 1 penetrations consist of a guard pipe anchored at the containment wall and welded to the flued head. The flued head is welded to the process pipe using a gradual buildup weld. The process pipe is allowed free axial thermal movement from the flued head through the drywell.

The guard pipe is allowed free axial thermal movement from the containment anchor point through its own sleeve at the drywell wall. Bellows, anchored to the drywell and welded to the guard pipe, will act as a seal for normal drywell environmental conditions. They are designed for thermal guard pipe expansion and relative seismic motion of guard pipe and drywell.

Type 2 penetrations consist of a penetration sleeve anchored in the containment and extending to just inside the liner. Full penetration welds are used to weld the flued head to the process pipe.

Type 3 penetrations consist of the sleeve anchored in the containment wall and extending just beyond the containment liner. Full penetration welds are used to attach the cover plate to the process pipe.

3.8.1.1.3.2 Electrical Penetrations

Dual header plate type electrical penetration assemblies are used to extend electrical conductors through the containment structure pressure boundary. These penetration assemblies are designed, fabricated, tested, and installed in accordance with the requirements of IEEE 317, "Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," dated December 1976.

Drawing E27-1310 and Figure 3.8-12 show electrical penetration plans, sections and details.

3.8.1.1.3.3 Personnel and Equipment Access Hatches

Two personnel access locks, shown in Figure 3.8-13, are provided for access to the interior of the containment.

Each personnel lock consists of an interlocked double door of welded steel assembly. Each door is equipped with a valve for equalizing pressure across the door such that the doors are not operable unless the pressure is equalized.

The two doors in each personnel lock are interlocked to prevent both being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened. An emergency lighting and communication system operating from an external auxiliary energy source is provided within the personnel locks.

The equipment hatch, Figure 3.8-13, is fabricated from welded steel and furnished with a double-gasketed flange and bolted dished door. The hatch barrel is welded to the containment liner.

Provisions are made to pressure test the space between the double gaskets of the door flanges. The weld seam tests channels at the liner joint and the dished door are provided to monitor any leakage during leak rate testing. Leak testing of the personnel hatches and the equipment hatch is discussed in Subsection 6.2.6.2.

3.8.1.1.3.4 Fuel Transfer Penetration

The inclined fuel transfer tube, shown in Figure 3.8-11 along with the three types of process pipe penetrations, penetrates the containment wall through the fuel transfer penetration. This is essentially a 3/4 inch thick carbon steel rolled plate pipe sleeve of 40-inch ID with a 36-inch standard flange on the containment side. The fuel transfer penetration forms a part of the containment boundary. Alternate isolation provisions for this penetration are decribed in Section 9.1.4.2.3.10.

3.8.1.1.4 Containment Liner

The containment wall liner is anchored to the wall with structural T sections. Typical wall liner anchorage details are shown in Figure 3.8-14. When a stiffener is cut to avoid interference with an insert assembly, welded studs are provided to restore anchorage of the liner plate.

Typical spacing of the liner anchors is 15 inches in the containment wall and the dome. Figure 3.8-15 shows details of dome liner anchors and stiffeners.

The top of the exposed base slab is lined with 1/2-inch and 1/4-inch stainless steel plate which serves as a leaktight boundary. The drywell wall and the sump floor are anchored through the base liner plate and into the base slab as shown in Figure 3.8-14. The spans of liner panels in the basemat area are:

a. pedestal cavity area: 3 feet 0 inch;

b. sump floor area: 6 feet 0 inch; and 20 feet 0 inch;

c. Suppression pool area: 3 feet 0 inch to 4 feet 8 1/4 inch (max.).

Leak test channels are provided at the liner seams in the suppression pool area and in the containment wall up to elevation 757 feet 0 inch. The containment liner in the wet areas of the suppression pool is of stainless steel to minimize corrosion problems.

3.8.1.1.5 Polar Crane Girder Brackets

The polar crane girder is located just below the spring line of the containment and is supported by brackets that are spaced 15° apart and embedded into the containment wall. Figure 3.8-16 shows the embedment details for the crane girder brackets.

3.8.1.2 Applicable Codes, Standards, and Specifications

This section lists codes, specifications, standards of practice, Regulatory Guides, 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 3.8-4 and are given with a specification reference number. The reference numbers for the containment are:

a. 1 through 5; 7 through 9;

b. 11 through 14, 16, 17, 18, 20, 21;

c. 23, 25, 28, 35, 36, 38 through 41;

d. 43, 44, 46, 47, 48, and 50.

Table 3.8-9 gives additional details regarding various codes used for design, material, fabrication, and erection of the major structural items within containment. Appendix B gives a detailed discussion on the construction material standards and quality control procedures required during construction.

3.8.1.3 Loads and Loading Combinations

The containment structure is designed using the loads, load combinations, and load factors listed in Table 3.8-1.1.

Loads and load combinations listed in Table 3.8-1.1 are used for the design of the steel liner and liner anchors, but the load factor for all load cases is 1.0.

Other steel elements serving pressure vessel functions, such as the hatches, locks, and pipe penetrations are designed for the loads and load combinations in Tables A3.9-6 and A-3.9-7. In addition to the loads defined in these tables, the pipe break loads are included under faulted conditions.

Structural steel elements, such as the polar crane brackets, are designed for the loads and load combinations in Table 3.8-2.

The seismic loads include the effects of both hydrostatic forces and hydrodynamic forces of the suppression pool water set in motion by seismic accelerations. Also, seismic loads are computed for both the one-and two-unit plant configuration.

Safety/relief valve discharge and LOCA related pool dynamic loads identified in Table 3.8-1.1 are discussed in Attachments A3.8 and A3.9.

The primary parameters used in the containment design are:

a. internal design pressure, 15 psig;

b. external design pressure, 3.0 psig;

c. calculated peak pressure, 8.74 psig;

d. test pressure, 17.25 psig; and

e. maximum suppression pool water temperature, 185°F.

Time-dependent pressure and temperature loads as discussed in Subsection 6.2.1 are simultaneously applied.

The effects of concrete volume changes are minimized by designing the concrete mix for minimum volume change (see Appendix B) and by prescribing construction procedure to minimize differential strains.

A one hundred year recurrence interval snow load of 25 psf has been considered in the design. This load is part of the dead load in the Load Combination Table 3.8-1.1, and part of the live load in Tables 3.8-1.2 and 3.8-2. For the extreme precipitation event (PMP) refer to Subsection 2.4.2.3. (Q&R 220.38)

3.8.1.4 Design and Analysis Procedures

3.8.1.4.1 General

The containment is analyzed using computer programs which are available in the Sargent & Lundy program library. These programs have all been validated by comparing result with selected problems where a closed-form solution is available or by comparing the solution of a given structure with the solution of the same structure obtained from one or more previously validated programs. These programs have been used very effectively on similar containments and have been found to be appropriate for containment analysis. A more detailed description of the various programs named in these subsections can be found in Appendix C.

Throughout the analysis, special attention is given to the following:

a. the intersection between the base slab and the cylinder;

b. the intersection between the cylinder and the dome;

c. the area around the large penetrations;

d. loading on the base slab from the underlying foundation material;

e. stresses due to transient temperature;

f. penetrations and points of concentrated loads; and

g. embedment of polar crane brackets in the containment wall.

3.8.1.4.2 Shell and Base Slab Analysis

The method of analysis used is a thin-shell of revolution finite element procedure using the computer program DYNAX. The complete containment with its basemat is modeled with shell elements.

The loads applied to the shell model are centerline loads. Consideration is given to the shift of the load from the actual place of application to the centerline of the shell. Overall effects of non-axisymmetric loads such as a pipe break load are analyzed using a series of Fourier harmonics, the summation of which represents the distribution of the load on the structure.

Results of the analysis, except for pool dynamic loads, are presented in Figure 3.8-17. Analysis results and design assessment of critical cross sections are presented in Attachment B3.8.

The base slab of the containment building is analyzed by a plate finite-element program PLFEM-II. The stiffening contributed by the walls is also included in the finite-element model.

Foundation soil is represented by equivalent springs at the nodal points of the basemat elements. A range of soil properties is used to allow for the short-term and long-term characteristics of the soil.

The base slab is also analyzed using the computer program CSEF III to confirm the results from the finite-element analysis.

The analytical models used for base slab analysis are shown in Figure 3.8-18.

3.8.1.4.3 Areas Around Large Penetrations

To determine the local effects around larger penetrations, such as the equipment hatch, main steam pipes, and personnel locks, the areas around these penetrations are modeled by the finite element program, PLFEM-II. The element nodes lie along the centerline of the containment wall, thus modeling the curvature of the wall. The size of the model is so chosen that the boundary conditions are compatible with those of an axisymmetric shell of revolution.

The areas around the large penetrations are designed for loads and loading combinations listed in Table 3.8-1.1.

3.8.1.4.4 Liner Analysis

Loads used for the analysis of steel liner plates are discussed in Subsection 3.8.1.3.

Force in a typical liner panel prior to buckling of any panel is determined from the net strain in the liner restrained by the surrounding containment wall. The liner anchorage system is modeled using the computer program LAFD, which calculates the post-buckling force and deflection of the anchors. The post buckling resistance of the panel is evaluated by the method outlined in Reference 1. This method is based on limit analysis and results in an upper bound for the anchor force and displacements. The anchor force-deflection functions are obtained from tests (Reference 2). The analytical method used to determine the post-buckling force in the liner and its anchor is described in detail in Reference 3.

The anchor is sized such that if failure were to occur it would be in the anchor and not the liner. The following cases are considered to produce the worst possible loading conditions on the anchorage system:

a. Case I - an initial inward deflection of 1/16 inch;

b. Case II - lower bound yield and 15% decrease in plate thickness of buckled panel;

c. Case III - upper bound yield and 15% increase in plate thickness in stable liner panels; and

d. Case IV - anchor spacing doubled to simulate failed or missing anchor. This case considers the post-buckling strength of this panel to be zero to maximize the load on the anchor.

3.8.1.4.5 Thermal Analysis

The containment is analyzed for thermal effects resulting from both operating and accident design conditions. The containment is designed using the loads and load combinations discussed in Subsection 3.8.1.3. Load combinations are used with and without the temperature loads, and the design is based on the critical case.

When considering the thermal effects, the steady-state gradients (an example of which is shown in Figure 3.8-19) are applied to the design section along with appropriate forces obtained from the containment analysis. The moments resulting from the thermal effects are permitted to change due to cracking of the concrete section. The stresses in the concrete and reinforcing steel are found by using the program TEMCO, which takes into account the extent of cracking of the section.

For the transient gradient, an equivalent linear gradient is found by summing moments about the centerline of the section. The section is then analyzed for the equivalent gradient by the same procedure used for the steady-state gradients.

3.8.1.4.6 Creep and Shrinkage Effects

Strains imposed by creep and shrinkage of the containment concrete are included in the design of the steel liner. In addition, minimum reinforcement is ensured throughout the containment structure to carry the effects of creep and shrinkage.

3.8.1.4.7 Suppression Pool Dynamic Load Analysis

The methods used for the analysis of safety-relief valve discharge and LOCA-related pool dynamic loads are described in Attachment A3.8.

3.8.1.4.8 Containment Ultimate Capacity

This section presents the details of a study to determine the ultimate containment capacity to withstand post-accident internal pressure.

3.8.1.4.8.1 Static Pressure Capacity and Associated Failure Mode

The containment is designed for an accident pressure of 15 psig along with appropriate concurrent loads and load factors presented in Table 3.8-1.1.

The calculated ultimate pressure capacity of the containment structure considering the liner as a load resisting element is 95 psig. If the liner is not considered for strength, the ultimate capacity of the containment shell and basemat is 75 psig. The upper and lower bound ultimate pressure capacities are based upon a statistical evaluation of mill test reports, as given in Table 3.8-8. For an analysis considering 2σ (σ = standard deviation) for the containment reinforcing and liner plate meterials, the ultimate pressure capacities are 95 ± 7.5 psig and 75 ± 5.3 psig with and without the liner, respectively. These ultimate pressure capacities correspond to initiation of yielding in the hoop reinforcement around the mid-height of the containment.

The ultimate capacity of the major containment penetrations is controlled by the equipment hatch. The static pressure capacity determined on the basis of the buckling of the equipment hatch spherical head is 76 psig. This was determined by using twice the basic allowable buckling stress, as specified in ASME Code Case N-284 Section 1400. To increase the margin of safety a factor of 1.2 can be applied to the analysis reducing the ultimate pressure retaining capability from 76 to 63 psig. Personnel air lock and equipment hatch door seals have been generically tested to 69 psig.

3.8.1.4.8.2 Design Basis

The design of the containment structure is based on ASME B&PV Code, Section III, Division 2, 1973 and the details are presented in Subsection 3.8.1.2. The design of the personnel lock and equipment hatch is based on ASME B&PV Code, Section III, Division 1, 1971 with addenda up to and including the Summer 1973 Addenda and the details are presented in Subsection 3.8.1.2.

3.8.1.4.8.3 Probable Failure Modes

Ultimate failure of the containment under internal pressure could result from one or a combination of the following:

a. For Pressure Boundary backed by concrete

1. Fracture of reinforcing steel

2. Punching of concrete around penetrations

b. For Pressure Boundary not backed by concrete

1. Fracture of steel components of penetrations such as equipment hatch and personnel locks, etc.

2. Buckling of the equipment hatch spherical head

3.8.1.4.8.4 Failure Criteria for Ultimate Capacity

For the purpose of this study, the failure of the containment is defined as attainment of any one of the following limits:

a. Stresses in reinforcing steel limited to yield stress;

b. strains in liner limited to the requirements of the ASME B&PV Code, Section III, Division 2, Subarticle CC3720, Factored Load Category;

c. stresses in portions of pressure boundary not backed by concrete limited to yielding of steel; or

d. critical buckling stress in the equipment hatch spherical head not to exceed twice the ASME Code allowable.

These limits are conservative and therefore there is reserve margin beyond the calculated ultimate pressure capacity.

3.8.1.4.8.5 Containment Finite Element Analysis

A nonlinear laminated shell finite element model was used to determine the ultimate pressure capacity of the containment. The Sargent & Lundy computer program DYNAX (see Appendix C for a description of the program) is used for the analysis. The program has the capability of analyzing reinforced concrete shells of revolution accounting for cracking of concrete and yielding of steel.

A sketch of the finite element model is shown in Figure 3.8-40 (for detailed engineering drawings, see Figures 3.8-3, 3.8-5, 3.8-6, and 3.8-14). The model uses 57 laminated shell elements and 58 nodes to represent the containment basemat, cylinder, and dome. Each shell element is represented as multiple concrete and steel layers. A sketch of the element layering in the sections is also shown in Figure 3.8-40. The number of layers included in any element varies from nine to thirteen. The layering of the elements allows the program to trace the non linear behavior and load distribution due to cracking of concrete and yielding of liner and reinforcement at each element under increasing pressure.

Actual mean material properties were used for the concrete, reinforcing steel, and steel liner. The actual material properties were established directly from the mill test reports and concrete cylinder tests for the material used in the construction of the containment structure. These properties are given in Table 3.8-8.

For the nonlinear analysis, the dead load of the structures, hydrostatic load on the suppression pool boundary, and the incremental internal pressure load were applied simultaneously. Even though the liner actually acts as a strength element, two sets of analyses were performed, one considering the liner as strength element and the other deleting the liner as strength element. The analysis traces the behavior of the entire structure at each pressure increment.

3.8.1.4.8.6 Results of Analysis

The results of the finite element analysis include stresses and strains in the steel reinforcing, concrete, and in the liner. These results show the overall axisymmetric response of the containment structure to incremental pressure.

Based on a review of the results, the following locations in the basemat, cylinder, and dome are identified as critical sections (see Figure 3.8-39):

a. Radial section in the basemat,

b. hoop section around the mid-height of containment, and

c. hoop section in the dome.

The most critical section was found to be the hoop section around the mid-height of containment where the yielding of reinforcement first occurs at a pressure of 95 psig with liner considered as strength element and at 75 psig when liner is not considered as strength element. At these calculated ultimate pressures, other sections in the containment are at reinforcement stresses which are less than yield and therefore the containment has some reserve margin beyond these calculated pressures.

Figures 3.8-41 through 3.8-45 summarize the containment response under increasing internal pressure loading, giving responses with and without the liner as a structural load carrying member. Figure 3.8-41 shows the variation of hoop reinforcing steel stresses in the most critical section around the mid-height of containment. As stated in Subsection 3.8.1.4.8.1, this is the failure mode of the containment which controls the ultimate capacity.

As seen from Figure 3.8-41, the hoop reinforcing stresses reach the average yield strength value of 71.1 ksi at 95 psig and 75 psig for the cases with and without the liner, respectively.

Also, as seen from Figure 3.8-45, the liner strain at 95 psig is 0.0025 in/in which is smaller than the strain allowed by ASME B&PV Code, Section III, Division 2, Subarticle CC3720, Factored Load Category.

3.8.1.5 Structural Acceptance Criteria

3.8.1.5.1 Reinforced Concrete

Deformations of the structure under factored loads are limited by specifying a maximum allowable concrete strain of 0.002 in./in. and by keeping the strains in the reinforcing steel not greater than the yield strain. However, the tensile strain in the reinforcing steel is allowed to exceed yield when the effects of thermal gradients through the concrete section are included.

For section analysis, the strain in the reinforcing steel and concrete is assumed to be directly proportional to the distance from the neutral axis. The concrete stress-strain relationship is defined by a half parabola whose apex is the point where the strain is 0.002 in./in. and the stress is 0.85 f'_c (f'_c being the specified concrete compressive strength). The tensile strength of the concrete is neglected.

Except for the allowable tangential shear stresses listed in Subsection 3.8.1.5.1.1, all reinforced concrete allowables are in accordance with Article CC 3400 of ASME B&PV Code, Sec. III, Div. 2.

3.8.1.5.1.1 Tangential Shear

The containment is designed for the peak tangential shear. The tangential shear stress capacity of concrete v_c is limited to 40 psi and 60 psi respectively for the service and factored load combinations defined in Table 3.8-1.1. The excess shear is designed to be carried by inclined reinforcement.

3.8.1.5.2 Steel Liner

The allowable stresses and strains for the liner plate are limited to values as specified in Article CC 3000 of ASME B&PV Code Sec. III, Div. 2. When subject to SRV discharge loads, the liner plates are designed in accordance with Subsection NE, Section III of the ASME B&PV Code.

As described in Subsection 3.8.1.5.2, the containment liner has been designed according to ASME Boiler and Pressure Vessel Code, Section III, Division 2, Article CC-3000. For SRV loadings, the liner in the suppression pool has been designed according to ASME Code, Section III, Division 1, Subsection NE requirements. The design also complies with the applicable provisions of Regulatory Guide 1.57. (Q&R 220.45)

3.8.1.5.3 Steel Pressure-Retaining Components

Portions of the containment boundary that are of steel and not backed by concrete, such as the equipment hatch, personnel locks and Code Class MC penetration assemblies including guard pipes, are designed in accordance with Subsection NE, Section III of the ASME B&PV Code.

These components are designed for the load combinations shown in Tables A3.9-6 and A3.9-7. The allowable stresses for these load combinations are summarized in the following list of figures from Section III Div. 1 of ASME B&PV Code:

a. design conditions, Figure NE-3221-1;

b. normal and upset conditions, Figure NB-3222-1;

c. emergency conditions, Figure NB-3224-1;

d. faulted conditions, Table F-1322; and

e. test conditions, Paragraph NE-3226.

3.8.1.5.4 Head Fitting Design

All head fittings (cover plates of flued heads), which are classified as Seismic Category I components, meet all stress requirements associated with the applicable design, operating, and testing conditions, as stated in the following paragraphs.

The allowable (temperature-dependent) stress values, as applicable to items a, b, c, d, and e, are taken from Tables I-1.1 through I-2.2 of the ASME Code, Section III, Div. 1.

a. Design Conditions

The head fittings are evaluated for design condition loadings and meet all applicable stress requirements set forth in Paragraph NB-3221 of the ASME Code, Section III. These requirements are shown on Figure 3.8-20. The design condition loading is the worst combination of the following loads:

1. design pressure and temperature,
2. weight loads,
3. operating base earthquake (OBE) loads, and
4. hydraulic transients.

b. Normal and Upset Conditions

The head fittings are evaluated for normal and upset condition loadings and meet all applicable stress requirements described in Paragraphs NB-3222 and NB-3223 of the ASME Code, Section III. These requirements are shown on Figure 3.8-21. The stress evaluation is conducted for the worst combination of the following loads:

1. operating pressure and temperature,
2. weight loads,
3. thermal expansion loads,
4. thermal and pressure transients,
5. OBE loads,
6. pool dynamic effects,
7. hydraulic transients, and
8. loads due to relative dynamic displacements.

c. Emergency Conditions

The head fittings are evaluated for emergency condition loadings and meet all applicable stress requirements set forth in Paragraph NB-3224 of the ASME

Code, Section III. These stress requirements are summarized in Figure 3.8-22. The stresses are evaluated for the worst combination of the following loads:

1. operating pressure and temperature,

2. weight loads,

3. SSE loads,

4. pool dynamic effects,

5. hydraulic transients.

d. Faulted Conditions

The head fittings are evaluated for the applicable faulted condition loadings and meet the stress requirements described in F-1324-1, F-1324.6, and Table F-1322 of Appendix F of the ASME Code, Section III, for system inelastic-component analysis. These stress requirements are summarized in Figure 3.8-23.

The stresses are evaluated for the following two loading cases:

1. Maximum Operating Pressures and Temperatures, plus loads due to Pipe Rupture and Jet Impingement, when applicable.

2. Process pipe maximum operating pressure applied in the annulus between the process pipe and the penetration sleeve for MC penetration assemblies only.

e. Testing Conditions

The head fittings are evaluated for test condition loadings in accordance with Paragraphs NB-3226, NB-6222, and NB-6322 of the ASME Code, Section III.

3.8.1.5.5 Penetration Sleeves and Guard Pipes

Containment penetration sleeve components including all guard pipes are Code Class MC and are designed and evaluated for design, operating, and testing conditions, in accordance with the following items, a, b, and c.

a. Design Conditions

Code Class MC penetration sleeves and guard pipes are evaluated for design condition loadings and meet all applicable stress requirements set forth in Paragraph NE-3221 of the ASME Code, Section III. These requirements are summarized in Figure 3.8-20, where the stress values, S_m, are the temperature-dependent allowable design stress intensity values. These values are taken from Table I-10.0 of Appendix I of the ASME Code, Section III. The stress evaluation is conducted for the loading combination described in item a of Subsection 3.8.1.5.4.

b. Operating Conditions

Code Class MC penetration sleeve components and guard pipes meet all applicable loading and stress requirements set forth in items b, c, and d of Subsection 3.8.1.1.3 for all operating conditions (normal, upset, emergency, and faulted), thus meeting the intent of Paragraph NE-3113 of the ASME Code, Section III.

c. Testing Conditions

Code Class MC penetration sleeve components are evaluated for testing conditions and satisfy the requirements specified in Paragraphs NE-6222, NE-6322, and NE-3131 of the ASME Code, Section III, for hydrostatic, pneumatic, or leak tests.

3.8.1.5.6 Basemat

In addition to the requirements listed in Subsections 3.8.1.5.1 and 3.8.1.5.2, the basemat meets the provisions of Subsection 3.8.5.5 with regard to bearing pressure, overturning, base sliding, and flotation.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

The construction materials and quality control procedures for the containment are specified in Appendix B.

Construction dimensional tolerances are specified on fabrication and construction drawings and in fabrication and construction specifications.

Criteria for establishing dimensional tolerances include:

a. fit-up, and

b. safety.

Fit-up of components is required to achieve high quality welded and bolted connections and desired interaction of abutting materials and components. Dimensional tolerances which ensure that geometry of completed structures is consistent with theoretical geometry used in design calculations are specified so that structures function safely under design loads.

Tolerance for out-of-roundness of the containment liner is specified in fabrication and construction specifications. This dimensional tolerance is set so that the containment vessel will function as a pressure vessel without introduction of stresses not considered in the design. Dimensional tolerances are set on locations of penetrations and embedment plates attached to the containment liner. These dimensional tolerances are consistent with construction techniques so that fit-up of the components is achieved.

3.8.1.7 Testing and Inservice Surveillance Requirements

3.8.1.7.1 Structural Acceptance Test

The structural acceptance test is performed after the containment is complete with liner, concrete structures, all electrical and piping penetrations, equipment hatch, and personnel locks in place.

The structural acceptance test is performed in accordance with Preoperational Test Procedure PTP-SIT-01, Revision 1, which meets the requirements of Article CC 6000 of ASME B&PV Code, Sect. III, Div. 2 (1980 Edition with Summer 1981 Addenda), with the exception that, because of the low design pressure, the pressure is brought up to 115% of the design pressure in only three increments, and tangential deflections at the equipment hatch are not measured because the deflections are negligible. At each pressure level the pressure is held constant for 1 hour before measuring the deflections at locations shown in Drawing S27-1401. The deflection is measured by taut wire extensometers stretched across the containment and kept under a constant tension.

At each pressure level, all cracks which exceed 0.01 inch in width and 6 inches in length are mapped at the following four locations:

a. near the base-mat/wall intersection,

b. midheight of the wall,

c. springline of the dome, and

d. equipment hatch penetration.

Table 3.8-6 shows the predicted deflections.

3.8.1.7.2 Leakage Rate Testing

Leakage rate testing is discussed in Subsection 6.2.6.

3.8.2 Steel Containment System

This subsection applies to the ASME Class MC Components of the concrete containment system described in Subsection 3.8.1. The MC components include the personnel and equipment access hatches, piping and electrical penetrations, and the fuel transfer penetration.

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 vessel support and shield the reactor, support recirculation pumps, support piping and auxiliary equipment, form the pressure suppression system, and provide pools and platforms for refueling operations. The internal structures include the following:

a. reactor shield wall,

b. drywell structure,

c. suppression pool weir wall,

d. reactor pedestal,

e. miscellaneous platforms and galleries,

f. containment pool,

g. refueling floor,

h. equipment rooms,

i. process pipe tunnel, and

j. support system for recirculation pumps.

3.8.3.1.1 Reactor Shield Wall

The reactor shield wall (Figure 3.8-25) is an open-ended cylindrical shell 2 feet 0 inch thick placed around the reactor pressure vessel. The primary function of the shield wall is to act as a radiation and heat barrier between the reactor pressure vessel and the drywell wall. It also provides support for pipes, pipe whip restraints, snubbers, and gallery work.

The shield wall consists of two concentric steel cylindrical shells, stiffened with radially placed diaphragms and filled with concrete in between the two shells. It is supported on top of the reactor pedestal ring girder (Figure 3.8-26).

Openings are provided for pipe penetrations and inservice inspection. The penetration sizes are minimized because inservice inspection is performed inside the shield wall. Since openings for inservice inspection are not in the high radiation area, shielding doors are not provided.

Additional stiffeners are provided, wherever necessary, for various attachments and around openings for local stiffening.

The shield wall is designed as a structural member to support equipment and piping loads as well as to resist pipe rupture, pressure, thermal, and seismic loads.

The presence of concrete inside the shield wall is neglected in determining the load-carrying capacity of the wall.

3.8.3.1.2 Drywell Structure

3.8.3.1.2.1 General Description

The drywell is a cylindrical reinforced concrete structure which surrounds the reactor pressure vessel and its support structure. The drywell is structurally designed as follows:

a. to provide structural support to containment pools, main steam tunnel and RWCU compartments;

b. to channel steam release from a LOCA through the horizontal vents for condensation in the suppression pool;

c. to protect the containment vessel from internal missiles and/or pipe whip;

d. to provide anchor points for pipes; and

e. to provide a support structure for the work platforms, monorails, pipe supports, and restraints that are located in the annulus between the drywell and the containment vessel.

The inside diameter of the drywell cylinder is 69 feet 0 inch, and the wall thickness is 5 feet 0 inch. The top of the drywell consists of a flat annular slab 6 feet 0 inch thick at elevation 803 feet 3 inches. The drywell wall is rigidly attached to the base slab at elevation 712 feet 0 inch (refer to Figure 3.8-14 for details). A steel head which can be removed to allow access to the reactor is located over the opening in the annular slab. Figures 3.8-1 and 3.8-2 show the drywell structure in plan and elevation.

The drywell is not normally entered during operation, but access is possible during a hot standby with the reactor subcritical.

The lower portion of the drywell wall is submerged in the suppression pool. Three rows of circular vents, 34 vents per row, penetrate the drywell wall below the normal level of the suppression pool. The surfaces of the drywell wall exposed to the suppression pool are lined with stainless steel clad plate 1 inch thick which is designed to act compositely with the drywell wall. Above the level of the suppression pool a carbon steel form plate 1/2 inch thick is provided on the interior surfaces of the cylinder walls and top slab. Structural T's and headed studs are attached to the form plate to provide mechanical anchorage of the plate to the concrete and to stiffen the liner for construction loads. The form plate provides a surface for forming the drywell walls and ceiling and minimizes bypass leakage, if any, through the drywell wall under accident conditions.

Details of the reinforcing in the drywell are shown in Figure 3.8-27. Reinforcing is deflected around small penetrations. At large penetrations additional bars are provided to account for concentration of stress. Reinforcing details around these penetrations are shown in Figures 3.8-28 and 3.8-29.

3.8.3.1.2.2 Drywell Penetrations

To maintain the drywell pressure boundary, drywell penetrations have the following design characteristics:

a. capability to withstand peak transient temperatures;

b. capability to withstand the forces caused by the impingement of the fluid from the largest local pipe or connection without failure;

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 the design pressure.

The number and sizes of the principal drywell penetrations are shown in Table 3.8-5. Figure 3.8-30 shows locations of these penetrations.

3.8.3.1.2.2.1 Pipe Penetrations

Piping penetrations are of the types used in the containment wall and are discussed in Subsection 3.8.1.1.3.1.

3.8.3.1.2.2.2 Suppression Pool Vents

There are 102 stainless steel-lined vent openings in three rows of 34 each around the base on the drywell (Figu re 3.8-30).

3.8.3.1.2.2.3 Electrical Penetrations

Drawing E27-1310 and Figure 3.8-12 shows a penetration of the general type that is used for the drywell wall.

3.8.3.1.2.2.4 Personnel and Equipment Access Hatches

Access to the drywell is provided by the drywell personnel lock, a personnel hatch located in the drywell ceiling, and the drywell equipment hatch shown in Figure 3.8-13. The personnel lock consists of an interlocked, double-door, welded steel assembly. Each door is equipped with a valve for equalizing pressure across the door such that the doors are not operable unless the pressure is equalized.

The two doors in the personnel lock are interlocked to prevent both being opened simultaneously, and to ensure that one door is completely closed before the opposite door can be opened. An emergency lighting and communication system operating from an external auxiliary energy source is provided within the personnel lock interior.

The equipment hatch is fabricated from welded steel and furnished with a double-gasketed flange and bolted, dished door. Provision is made to pressure test the space between the double gaskets of the door flanges. A shield wall is provided with the same shielding requirements as the drywell wall.

3.8.3.1.2.2.5 Access for Refueling Operations

The drywell head (Figure 3.8-31) is removed during refueling operations. This head is held in place by bolts and sealed with a double seal. It is opened only when the primary coolant temperature is below 212°F and the core is sub-critical. The double seal provides a method for determining the leak tightness of the seal without pressurizing the drywell.

3.8.3.1.3 Suppression Pool Weir Wall

The suppression pool weir wall, located inside the drywell, acts as the inner boundary of the suppression pool. It is constructed of reinforced concrete and extends from the outer edge of the drywell sump floor. The weir wall is lined with 1/4-inch stainless steel plate on the suppression pool side to protect the concrete from demineralized water. Vertical angles 3 inches x 3 inches x 3/8 inch spaced at 15 inches on center are used to stiffen and anchor the weir wall liner.

The weir wall is reinforced (Figure 3.8-27) 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 as required for transverse shear.

The principal dimensions of the weir wall are:

a. Inside diameter: 61 feet;

b. Wall thickness: 1 foot 10 inches;

c. Height above basemat: 23 feet 9 inches; and

d. Height above sump floor: 12 feet 7 1/4 inches.

3.8.3.1.4 Reactor Pedestal

The reactor pedestal (Figure 3.8-26 and 3.8-32) supports the reactor pressure vessel (RPV) and reactor shield wall. The pedestal shell is a steel structure consisting of two concentric cylindrical shells connected by radially placed steel diaphragms for the entire height of the cylinders. The top of the pedestal consists of a ring girder to which the reactor shield wall is welded. The RPV base is anchored to the ring girder by pretensioned bolts which are designed to carry the loads through friction. Openings are provided through the pedestal for access, control rod drive piping, and nuclear instrumentation. To increase the stability of the structure, the annulus between the steel cylinders is filled with concrete. The concrete is not considered to act compositely with the steel plates.

The base of the pedestal is welded to embedded plates anchored in the sump floor with reinforcing bars attached to the plates (Figure 3.8-26).

The principal dimensions of the pedestal are:

a. outside diameter of top ring girder: 29 feet 10 inches;

b. inside diameter of top ring girder: 16 feet 6 inches;

c. outside diameter of outer steel shell: 29 feet 10 inches;

d. inside diameter of inner steel shell: 18 feet 6 inches; and

e. height above basemat: 31 feet 2 inches.

3.8.3.1.5 Miscellaneous Platforms and Galleries

Miscellaneous platforms and galleries inside the containment serve the dual function of providing access to the electrical and mechanical equipment and providing structural support for this equipment. The platforms and galleries consist of either concrete slabs cantilevered from the drywell wall or structural steel framing supported on containment, drywell, reactor pedestal, and shield walls. Thermal loads in the gallery framing are considered for those beams where thermal expansion is a concern. The layout and configuration of the framing is such that no significant radial thermal loads are imposed on any of the walls.

3.8.3.1.6 Containment Pool

The containment pool supported on the drywell walls has the following functions:

a. to provide shielding when the reactor is in operation;

b. to provide storage space for the dryer and separator assemblies; and

c. to provide an area for fuel transfer during refueling.

This pool forms a rectangular box across the top of the drywell, which is integrated into the design of the top of the drywell. The weight of the pool is transmitted to the foundation mat through the drywell walls. The interior of the pool is lined with stainless steel plate.

3.8.3.1.7 Refueling Floor

The refueling floor provides laydown space for reactor components and refueling equipment. The concrete portion of the floor is designed as an integral part of the drywell structure. The grating portions of the floor are supported by structural steel framing which is supported by the containment pool walls and the containment walls. Connections to the containment wall are designed to transfer only vertical and lateral reactions to the containment structure.

3.8.3.1.8 Equipment Rooms

Equipment rooms, located near the top of the drywell, are constructed of reinforced concrete. They are designed as an integral part of the drywell structure and are not supported by the containment walls. The rooms are provided with openings that connect directly to the containment volume. The roof of equipment rooms also is part of the refueling floor.

3.8.3.1.9 Process Pipe Tunnel

The process pipe tunnel provides shielding for the process piping between the drywell and the containment. It is designed as an integral part of the drywell structure and is constructed of reinforced concrete. The arrangement at the containment wall permits differential movement between the tunnel and the containment. Doorways connect the tunnel to the containment volume.

3.8.3.1.10 Drywell Sump Floor

The drywell sump floor is a thick slab of reinforced concrete which rests on the basemat and supports the suppression pool weir wall and the reactor pedestal. It is anchored through the containment liner to the basemat with reinforcing bars, as shown in Figure 3.8-14. A stainless steel liner is provided on the suppression pool side to protect the concrete from demineralized water.

The sump floor is reinforced with bars placed in the radial, vertical and hoop directions. The reinforcement details are shown in Figure 3.8-27.

The sump floor has the following principal dimensions:

a. inside diameter: 18 feet 6 inches:

b. outside diameter: 64 feet 8 inches; and

c. thickness: 11 feet 1-3/4 inches.

3.8.3.1.11 Support System for Recirculation Pumps

The recirculation pump and motor assemblies lie above elevation 729 feet 8 inches in the drywell on opposite sides of the reactor pedestal. The pump and motor assemblies are supported by four constant-support spring hangers which attach to built-up box girders at the top. The box girders form a lattice configuration with radial members spanning between the shield and drywell walls, and tangential members bearing on top of the radial - members. Seven snubbers and two struts are attached to the pump and motor to protect it against dynamic loads.

The two struts are attached to the reactor pedestal, three of the snubbers are attached to the reactor shield wall, and the other four snubbers are attached to embedded plates on the drywell sump floor at elevation 723 feet 1 3/4 inch.

3.8.3.2 Applicable Codes, Standards, and Specifications

This subsection lists codes, standards of practice, regulatory guides, 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 3.8-4 and given a reference number. For each structure internal to the containment, the reference numbers are listed in Subsections 3.8.3.2.1 through 3.8.3.2.4.

Table 3.8-9 gives additional details regarding various codes used for design, material, fabrication, and erection of the major structural items within containment. Appendix B gives a detailed discussion on the construction material standards and quality control procedures required during construction.

3.8.3.2.1 Reactor Shield Wall and Pedestal

The reference numbers are as follows:

a. 25b for reactor pedestal;

b. 21 and 23 for reactor shield wall; and

c. 28, 41, 43, and 47 for both reactor pedestal and reactor shield wall.

3.8.3.2.2 Drywell Structure

The reference numbers are as follows:

a. 1 through 5; 7 through 9;

b. 11 through 14, 16, 17, 18, 20;

c. 21, 23, 25a, 28;

d. 35, 38, 41, 43, 44, 46, 47, 48, and 49.

The CPS design is in compliance with ACI 349-76 and Regulatory Guide 1.142 with the following clarifications:

a. Requirements of Section 10.6.3 of ACI 349 is met as given in ACI 349-80.

b. Regarding the bend test requirements for bar nos. 3 through 11, CPS has followed ASTM A615 requirements, which specify slightly bigger bend radii than ACI 349. The following tables gives a comparison of ACI 349 vs. ASTM A615 requirements:

Bar Designation Numbers	ACI 349 Radius	ASTM A615 Radius
3, 4, 5	3 1/2 d	4 d
6	5 d	5 d
7, 8	5 d	6 d
9, 10, 11	7 d	8 d

It should be noted that the primary purpose of the bend test is to assure against cracking during the bending of bars. However, cracking is related directly to the ductility of bars. The ductility of bars at CPS exceeded significantly those required by ACI 349 and ASTM A615. A comparison of the required elongation vs. the actual statistical average elongation, as determined from the Certified Mill Test Reports, is given below:

Bar Designation Number	Required Elongation (Percent)	Actual Statistical Average Elongation (Percent)
5	9	12.03
6	9	10.86
7	8	14.44
8	8	12.47
9	7	14.17
10	7	15.10

Bar Designation Number	Required Elongation (Percent)	Actual Statistical Average Elongation (Percent)
11	7	16.17
14	7	15.56
18	7	14.54

Based on the above comparison, the effect of the minor difference in the bend radii requirement, if any, is offset by the much higher elongation of the bars used at CPS. (Q&R 220.49)

3.8.3.2.3 Miscellaneous Platforms and Galleries, Refueling Floor, Equipment Rooms, Suppression Pool Weir Wall, Process Pipe Tunnel, and Structural Support System for Recirculation Pumps

The reference numbers are as follows:

a. 1 through 5, 7 through 9;

b. 11 through 14, 16, 17, 18, 20;

c. 21, 23, 28, 31, 35;

d. 37, 38, 41, 43, 46, 47, and 48.

3.8.3.2.4 Containment Pool

The reference numbers are as follows:

a. 1 through 5, 7, 8, 9;

b. 11 through 14, 16, 17, 18, 20;

c. 21, 23, 25d, 28, 35;

d. 37, 38, 41, 43, 44, 46, 47, and 48.

3.8.3.3 Loads and Loading Combinations

The reinforced concrete internal structures, which include the drywell, suppression pool weir wall, containment pool, equipment rooms, process pipe tunnel, and portions of the refueling floor, are designed using the loads, load combinations, and load factors listed and discussed in Table 3.8-1.1. In the vent area of the drywell, the steel liner is well anchored to the concrete to ensure composite action and is designed using Table 3.8-1.1.

The weir wall, the lower portion of the drywell, and the containment pool are designed to resist the effects of both hydrostatic forces and hydrodynamic forces associated with water set in motion by seismic accelerations.

The drywell is designed for the differential pressure between the drywell and containment as discussed in Subsection 6.2.1. Time-dependent pressure and temperature loads as discussed in Subsection 6.2.1 are simultaneously applied.

The upper containment pool is designed for a maximum water temperature of 212° F. The weir wall and the portion of the drywell located inside the suppression pool are designed for a maximum water temperature of 185° F.

Safety/relief valve discharge loads and LOCA-related pool dynamic loads identified in Table 3.8-1.1 are described in Attachments A3.8 and A3.9.

The structural steel elements of the internal structures, which include the reactor pedestal, the reactor shield wall, portions of the refueling floor, the miscellaneous platforms and galleries, and the structural support system for recirculation pumps are designed using the loads and load combinations listed and discussed in Table 3.8-2.

The platforms in the annulus between the drywell and the containment are subjected to upward loads from pool swell, P, and seismic effects. These three upward-acting loads form the basis for the design.

The differential pressure (P) and pool swell forces are combined with live, dead, and seismic loads as applicable.

The thermal loads associated with the reactor shield wall include temperature gradients under normal operating and accident conditions due to the absorption of gamma and neutron radiation.

Internal structures are designed for the reactions of all other structures or equipment that they may support.

The effects of concrete volume changes are minimized by designing the concrete mix for minimal volume changes (see Appendix B) and by prescribing construction procedures to minimize differential strains.

Closed compartments are designed to withstand the temperatures and pressures due to the failure of equipment which is inside of the compartments.

The reactor shield wall is designed for loads resulting from pipe breaks within the RPV shield wall annulus. These loads are described in Subsection 6.2.1.2.1.2.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 Reactor Shield Wall

The reactor shield wall is designed as a cylindrical shell with inner and outer shell plates stiffened by closely spaced vertical stiffener plates. The concrete filling acts mainly as a radiation shield and is not considered as a structural element to carry any load.

Two computer programs, DYNAX and SLSAP-4, are used for the analysis of the reactor shield wall. The shield wall is analyzed as a symmetrical shell in the DYNAX run mainly to calculate stresses resulting from thermal and pressure loads. The reactor shield wall is analyzed as an anisotropic shell for the effects of pipe rupture loads. The finite-element computer program SLSAP-4 is used for this analysis. Appropriate boundary conditions are used at the base connection with the reactor pedestal for both the models.

The pipe rupture loads are applied as concentrated loads at the node points of the three-dimensional finite-element model.

The stresses due to different loadings are taken from respective computer output and combined manually to arrive at the final design stress.

3.8.3.4.2 Drywell and Attached Structures

3.8.3.4.2.1 General

The drywell is treated as a structure with a cylindrical wall and an annular roof, with the containment pool walls, process pipe tunnel and equipment rooms rigidly attached. The steel plates in the vent area are anchored to the drywell wall to ensure composite action and are designed to transfer loads to the foundation.

Throughout the analysis, special attention is given to the following:

a. the intersection between the base slab and the cylinder;

b. the intersection between the cylinder and the roof;

c. the intersection between the pool walls and the roof or cylinder;

d. the stresses around penetrations;

e. the stresses caused by transient temperatures in steel liners and concrete; and

f. penetrations and points of concentrated loads.

3.8.3.4.2.2 Shell Analysis

The drywell is divided into two portions for analysis, with the match line located near the mid-height of the cylinder at elevation 755 feet 0 inch. A simplified model of the drywell cylinder wall and slab which includes the weight of the compartment walls and slabs and pool walls was analyzed to obtain results in the lower portion. The upper portion is analyzed by applying boundary conditions at the match line such that compatibility of deformations and equilibrium of forces is maintained.

Analysis of the upper portion of the drywell utilizes the computer program SLSAP-4 to determine the distribution of forces. Taking advantage of the symmetry of the structure about the east west axis, half of the upper drywell, including the fuel pools and equipment compartments, is modeled in Figure 3.8-34.

Results in the lower portion of the drywell are obtained using the computer programs DYNAX and PLFEM-II. DYNAX is used to analyze the axisymmetric model for both axisymmetric and non axisymmetric loadings. Non-axisymmetric loads are approximated by Fourier series. The effect of discontinuities, e.q. around large penetrations, is considered by analyzing portions of the structure using PLFEM-II.

The effect of the vents in the suppression pool area is taken into account by modifying the shell stiffness to obtain the force resultants in the area. The stress distribution and the vents are then more closely analyzed for stress concentration using PLFEM-II.

Pressure loads applied to the models are centerline loads. Therefore, consideration is given to the shift of the load from the actual place of application to the centerline of the shell.

The results of the analysis except for pool dynamic loads are shown in Figure 3.8-33. Analysis results and design assessments of critical cross sections are presented in Attachment B3.8.

3.8.3.4.2.3 Areas Around Large Penetrations

To determine the local effects at large penetrations, such as the equipment hatch and main steam pipes, the areas around these penetrations are modeled by a finite-element program, PLFEM-II. The element nodes lie along the centerline of the drywell wall, thus modeling the curvature of the wall. The size of the model is so chosen that the boundary conditions are compatible with those of an axisymmetric shell of revolution.

3.8.3.4.2.4 Thermal Analysis

The drywell is analyzed for thermal effects resulting from both operating and accident design conditions. The drywell is designed using the loads and load combinations discussed in Subsection 3.8.3.3. Load combinations are used with and without the temperature loads, and the design is based on the critical case.

When considering the thermal effects, the steady-state gradients are applied to each design section along with concurrent axial loads and moments in a design load combination. The stresses in the concrete and reinforcing steel are then analyzed using the computer program TEMCO, which considers the extent of cracking of the section and the resultant change in the thermal moment. 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.

3.8.3.4.2.5 Creep and Shrinkage Effects

Sufficient reinforcing is provided throughout the drywell structure to carry the effects of creep and shrinkage.

3.8.3.4.2.6 Suppression Pool Dynamic Load Analysis

The drywell structure is analyzed for the SRV discharge loads and LOCA related pool dynamic loads as described in Attachment A3.8.

3.8.3.4.3 Reactor Pedestal and Suppression Pool Weir Wall

The reactor support pedestal and the suppression pool weir wall are designed as axisymmetric cylindrical shells fixed at their base. Loads from the reactor pressure vessel and the reactor shield wall are applied at the top of the pedestal. Two Sargent & Lundy shell-of-revolution programs, SOR-III and DYNAX, are used for analysis. The seismic and pipe rupture forces transmitted to the pedestal are included in the design as shears and overturning moments.

The effects of concentrated pipe break load on the weir wall are analyzed using a series of Fourier harmonics, the summation of which represents the distribution of the load on the structure. The capacity of the section under combined loads is checked using the program TEMCO. Thermal analysis is performed as discussed in Subsection 3.8.3.4.2.4. The weir wall thickness and reinforcement are so proportioned that the pressure suppression efficiency is not impaired by the deflection of the weir wall under design loads.

3.8.3.4.4 Refueling Floor, Miscellaneous Platforms and Galleries, and Support System For Recirculation Pumps

The platforms, galleries, and structural supports for the recirculation pumps are designed using conventional elastic design methods.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Reinforced Concrete

Deformations of the drywell structure, containment pools, and equipment rooms under factored load conditions are limited by specifying a maximum allowable concrete strain of 0.002 in./in. Yielding of the reinforcing steel in tension is allowed only when the effects of thermal gradients are considered.

For section analysis, the strain in the reinforcing steel and concrete is assumed to be directly proportional to the distance from the neutral axis. The concrete stress-strain relationship is defined by a half parabola whose apex is the point where the strain is 0.002 in./in. and the stress is 0.85 f_c, where f_c is the specified concrete compressive strength. The tensile strength of the concrete is neglected.

Except for the allowable tangential shear stresses listed in Subsection 3.8.3.5.1.1, reinforced concrete allowables for the drywell structure, containment pools, equipment rooms, and sump floor and weir walls are in accordance with ASME B&PV Code, Section III, Div. 2.

The stresses and strains in all other reinforced concrete internal structures are limited to those specified in ACI 318. Serviceability checks are made in accordance with ACI 318 to assure crack control and to keep deflections below the limits prescribed in ACI 318 or to the manufacturer's recommendations for equipment supported by the reinforced concrete. The factors of safety against material strength are contained in the load factors in Table 3.8-1.2 and in the capacity reduction factors (θ) in ACI 318.

3.8.3.5.1.1 Tangential Shear for Drywell

The allowable tangential shear stress for concrete, under the factored load conditions defined in Table 3.8.1-1, is calculated in accordance with Section 11.16 of ACI 318. The shear stress carried by concrete under service load conditions is 50% of the value allowed for the factored load conditions.

Reinforcement is provided to carry the shear in excess of the concrete shear capacity in accordance with Section 11.16 of ACI 318 except that for service load conditions, 50% of f_y is used.

The original drywell design for tangential shear is based on Section 11.16 of ACI-318. This question was also addressed in CPS-FSAR Amendment 16, dated July 6, 1974, in response to Question 3-71, Page 3-120 (see below). Tangential shear design for the drywell is also in compliance with Section 11.10 of ACI-349 which the NRC is currently accepting, as stated in Question 220.49.

Structural acceptance criteria for drywell stated in Subsection 3.8.3.5.1 is in agreement with the criteria given in SRP, Section 3.8.3.II.5, with the exception of tangential shear as discussed above.

The drywell is different from the containment structure and therefore it is considered that Sections CC-3411-5 and CC-3521-1 of the Containment Code, ACI-ASME 359, do not apply to the drywell structure. Criteria in Section 11.16 of ACI-318 code will be used in the design of the drywell and reactor pedestal. (Q&R 220.51)

3.8.3.5.2 Structural Steel

The stresses in the structural steel are limited to those specified in the AISC Specification, Part 1, when designing for the loading combinations in Table 3.8-2, combinations 1, 2, and 4 through 8. For loading combination 3 of Table 3.8-2, the allowable steel stresses are increased to 1.33 time those specified in the AISC Specification. The appropriate factors of safety against yield are as discussed in the Commentary to the AISC Specifications. The allowable steel stresses are increased to 1.6 times those specified above, subject to an upper bound of 0.95 F_y (yield stress), when designing for the loading combinations in Table 3.8-2, combinations 9 through 18. In this situation a minimum factor of safety of 1.0/.95 = 1.05 against yield will be assured. In both cases, deformation of the steel members is limited by keeping the steel stresses within the elastic range for all loading combinations that exclude the effects of LOCA loads. When LOCA loads are considered in the loading combination, local stress may go beyond yield as long as overall function of the structure is not impaired. Connection parts may be allowed to yield locally under abnormal and/or extreme environmental loading conditions as long as the overall capacity is shown to be adequate.

3.8.3.5.3 Suppression Pool Liner Plate

The allowable stresses for the suppression pool liner plate in the vent area of the drywell are those specified for reinforcing steel in Article CC 3000 of ASME B&PV Code, Section III, Div. 2.

3.8.3.5.4 Steel Pressure-Retaining Components

Portions of the drywell boundary that are not backed by concrete, such as the equipment hatch, personnel lock, drywell head, guard pipes for the eleven high energy process lines, and MC penetration sleeves are designed in accordance with Subsection NE of Section III of the ASME B&PV Code.

These components are designed for the loads and load combinations shown in Tables A3.9-6 and A3.9-7. In addition to the loads defined in these tables, the applicable pipe break loads are included under faulted conditions. The allowable stresses for these load combinations are summarized in the following list of figures from Section III of ASME B&PV Code:

a. design conditions, Figure NE-3221-1;

b. normal and upset conditions, Figure NB-3222-1;

c. emergency conditions, Figure NB-3224-1;

d. faulted conditions, Table F-1322; and

e. test, Paragraph NE-3226.

3.8.3.5.5 Reactor Pedestal Steel

The stresses in the pedestal shell plates and diaphragms are limited to those specified in Figure NF-3221-1 of Subsection NF (components and supports) of ASME B&PV Code Section III, Division 1.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

There is no danger of radiation damage to the structures internal to the containment because, except for the reactor shield wall, they are not in a region of high-energy neutron flux. There is no danger of radiation damage to the steel plates of the reactor shield wall, because damage occurs at a neutron fluence of about 10^{22} nvt. It has been determined that in the 40-year life expectancy of the station, the inside face of the wall will experience a neutron fluence of less than 5×10^{17} nvt.

The construction materials and quality control procedure for all concrete and structural steel internal structures conform to the standards set forth in Appendix B.

3.8.3.7 Testing and Inservice Surveillance Requirements

After the drywell is complete with liner, concrete and steel structures, electrical and piping penetrations, equipment hatch, and personnel lock, it is tested in accordance with the procedure outlined for a drywell in SRP 3.8.3.7.

The drywell is pressurized to the design pressure of 30 psig in four approximately equal increments. At each pressure level, the pressure is held constant for 1 hour before measuring the strains and deflections at locations shown in Drawing S27-1401. The deflection is measured by taut wire extensometers stretched across the drywell and kept under a constant tension. Electrical strain gauges are used to measure the strains through the thickness of the drywell wall.

The structural integrity of reinforced concrete members is also evaluated by mapping cracks larger than 0.01 inch in critical areas identifiable by design.

Table 3.8-7 shows the predicted deflections of the drywell during the pressure test. The acceptance criteria for the drywell are the same as outlined for the primary reactor containment (see Subsection 3.8.1.7.1).

3.8.4 Other Seismic Category I Structures

3.8.4.1 Description of the Structures

The Seismic Category 1 structures, excluding the containment and containment internal structures, are:

a. auxiliary building,

b. fuel building,

c. control building,

d. diesel generator and HVAC (DG & HVAC) building,

e. radwaste building substructure,

f. containment gas control boundary,

g. portions of circulating water screen house (CWSH), and

h. UHS discharge structure.

Refer to Section 1.2 for general arrangement drawings which show the relative size and location of the above structures.

3.8.4.1.1 Auxiliary Building

The reinforced concrete auxiliary building is located adjacent to and fits the contour of the containment building on one side. This structure is supported on a mat foundation which is continuous with the mats under the containment structure, the control building, and the turbine building. Above the foundation mat, the auxiliary building is structurally isolated from the containment structure, but structurally connected to the control and turbine buildings.

The main steam tunnel extends from the containment to the turbine room through the auxiliary building. It houses the process piping and protects it from the effects of external missiles. In the unlikely event of pipe rupture inside the tunnel, it protects the control room and other Seismic Category I equipment and components from the effects of radioactive steam and pipe rupture loads. The tunnel also provides supports and restraints for the process piping.

The ECCS pump rooms, in the lowest level of the auxiliary building, are in flood protection compartments with watertight doors. In the event of a pipe rupture, the flooding in one compartment will not result in the flooding of any other compartment, and the failure of a pump suction line will not drain the suppression pool. The second or grade level of the auxiliary building houses pump room access hatches and a cable tunnel. The third and fourth levels are provided for electrical switchgear and electrical penetrations.

3.8.4.1.2 Fuel Building

The fuel building fits the contour of the containment structure on one side and is adjacent to the DG and HVAC building. One side of the building accommodates the fuel shipping cask railroad car or transporter used for dry cask storage operations. The reinforced concrete structure is supported on a concrete mat which is continuous with the mats under the containment structure and the DG and HVAC building. The fuel building above the mat is structurally isolated from the containment building, but structurally connected to the DG & HVAC and auxiliary buildings.

The three pools in the fuel building provide for fuel transfer, spent fuel storage, and cask loading. The pools are lined with seam-welded stainless steel plate welded to reinforced members embedded in the concrete. Channels are located behind the weld seams of the pool liners and are monitored to detect possible leakage from the pools. The reinforced concrete construction above the main floor provides missile and tornado protection.

Crane seismic safety features, Figure 3.8-35, are provided for the fuel building crane. The jurisdictional boundary between components of the crane system is described in USAR Figure 3.8-35. The crane is evaluated to comply with NUREG-0554 as described in USAR Section 9.1.4.2.2.2. The requirements of USAR Section 3.8.4.5 do not apply to the rails, rail clips, and rail clip to girder bolts.

3.8.4.1.3 Control Building

The reinforced concrete control building is located next to the auxiliary building. It is also adjacent to the DG & HVAC and radwaste buildings. The control building is supported by a reinforced concrete basemat which is continuous with the basemats of the adjoining buildings. The building is structurally connected to the adjoining buildings above the basemats.

HVAC equipment is located in the lower two and the top levels of the control building. Laundry facilities and laboratories are located at the grade or third level. The component cooling water heat exchangers are on the fourth level. The fifth level houses switchgear and provides a cable spreading area. The control room is located on the sixth level.

The reinforced concrete walls and slabs of the control building provide tornado and missile protection for the control room and other Seismic Category I systems.

3.8.4.1.4 Diesel Generator and HVAC Building

The reinforced concrete DG & HVAC building is located next to the fuel building and adjacent to the control building. The DG & HVAC building is supported by a reinforced concrete base mat which is continuous with the basemats of the adjoining buildings. The building is structurally connected to the adjoining buildings above the basemat.

The fuel oil storage tanks are located on the lower level, and diesel generators are located at grade level. HVAC equipment is located above the grade floor. The reinforced concrete construction provides tornado and missile protection.

3.8.4.1.5 Radwaste Building Substructure

The reinforced concrete radwaste building is located adjacent to the turbine building and the control building. The radwaste building is supported by a concrete basemat which is continuous with the basemats supporting the turbine building and the control building.

The building is structurally connected to the adjoining buildings.

The radwaste building houses station systems which are required to process and dispose of radioactive wastes generated during power operation. The reinforced concrete construction prevents the dispersion of waste material by tornadic winds. Only the portion of the radwaste building located below grade is designed as a Seismic Category I structure.

3.8.4.1.6 Containment Gas Control Boundary

The containment gas control boundary is a limited leakage structure which surrounds the containment structure above the auxiliary and fuel buildings. The enclosure conforms to the shape of the containment and is separated from it by a distance of approximately 4 feet. The enclosure is made up of siding supported by structural steel framing attached to the containment.

3.8.4.1.7 Circulating Water Screen House

The circulating water screen house (CWSH) is a reinforced concrete, Seismic Category I structure located northwest of the station site. The CWSH is supported by a reinforced concrete basemat. The substructure is constructed of reinforced concrete. The shutdown service water pump cubicles in the superstructure are of reinforced concrete, but the rest of the superstructure is constructed of structural steel.

The CWSH houses the plant service water pumps and strainers, the shutdown service water pumps and strainers, the diesel-driven fire pump and associated equipment, the circulating water pumps, and the traveling screens. All pumps are the vertical wet-pit type. The shutdown service water system equipment is the only equipment in the CWSH that is required to safely shut down the reactor or to maintain it in a safe shutdown condition. The three shutdown service water pumps and strainers are in their own missile-protected cubicles. Each cubicle has its own cooling unit which is electrically segregated from the others. Each cubicle is flood protected by bulkhead doors. No single failure of the equipment associated with one cubicle will have a detrimental effect on the rest of the system. The CWSH basin is constructed with two inlet channels to provide water to the shutdown service water pumps.

3.8.4.1.8 Ultimate Heat Sink Discharge Structure

A reinforced concrete structure is located at the ultimate heat sink to accomodate the shutdown service water discharge lines.

3.8.4.2 Applicable Codes, Standards and Specifications

The codes, standards and specifications applicable to the design, fabrication, construction, testing and in-service inspection of safety-related structures outside the containment are listed in Table 3.8-4 and include the following specification numbers:

a. 1 through 5, 7, 8, 9, 11 through 14, 16, 17, 18, 20;

b. 21, 23, 25c, 25d, and 28;

c. 31, 35, 37, 38;

d. 41, 43, 44, and 46 through 48.

3.8.4.3 Loads and Loading Combination

The list of loads and their definitions and the loading combinations applicable to the design of Seismic Category I structures outside the containment are given in Table 3.8-1.2 and 3.8-2. The list of load categories where the types of loads are defined is also given 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 piece of equipment. The roofs are designed for a uniform live load of 25 psf in addition to snow loads and loads from probable maximum precipitation. The roofs are also designed to withstand suction pressure induced by the design wind and tornadic wind as discussed in Section 3.3. Pattern live loads are applied to determine maximum moments and shears in each slab. All slabs are designed for the effects of internal missiles, thermal gradients, and pipe rupture loads, wherever applicable. Floors and roofs are checked for their ability to transfer shear through diaphragm action.

The walls, interacting with the floor slabs, are designed to withstand the effects of seismic induced shears and moments. All walls are designed for external and internal missiles, transient thermal gradients, tornado-induced pressure, lateral soil and hydrostatic pressure and pipe rupture loads, wherever applicable, in addition to their own weight and associated loads from slabs and beams framing into the walls. For the design of subgrade walls a surcharge load of 500 psf, 1000 psf for E-70 or 300 psf for AASHO H-20 wheel loading is considered.

The CWSH is also designed for the hydrostatic and hydrodynamic effects of the cooling lake water.

The pools in the fuel building are designed for, in addition to applicable loads listed above, hydrostatic loads and hydrodynamic loads associated with water set in motion by seismic accelerations. The pools are designed for the effects of a maximum water temperature of 212°F.

The containment gas control boundary is designed to be held under a negative pressure equivalent to 1/4-inch of water when infiltration flow rates given in Subsection 6.2.3 are being passed through the standby gas treatment system. The containment gas control boundary is not designed to withstand the effects of missiles. The siding of the enclosure is designed to fail for wind speeds of 200 mph, which is less than the design basis tornado. The gas control boundary is a fission product barrier only, and it is not designed for the high temperatures and pressures which are postulated for the containment. The steel framing for the containment gas control boundary is designed to withstand effects of tornado loading.

In all instances, the Seismic Category I structures and structural components are designed for the vertical and horizontal accelerations associated with both SSE and OBE.

In Radwaste, Control and Diesel Generator Buildings, the effects of pool dynamic loads associated with SRV actuation and LOCA are considered negligible and shall not be used for the analysis and design of these buildings.

The Category I manholes, buried piping, electric ducts and tunnels were designed for both dead and live load, and seismic loading conditions. In addition, the buried piping was designed for thermal expansion. The manholes, electrical ducts and tunnels are designed using the strength design method. The buried piping is designed using the ASME Section III (1977) stress equations.

The design procedure complies with the criteria contained in SRP, Section 3.8.4. (Q&R 220.54)

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 significant openings and discontinuities in structural members are included in the structural model. The boundary conditions selected for all structural models are determined by evaluating the stiffnesses (flexural, torsional and axial) of all the members connected to a boundary point and represent, to the extent practical, the actual restraint conditions.

The walls, interacting with the floor slabs, are proportioned to resist the combination of seismic-induced overturning moments, vertical loads, and shears in accordance with the special provisions for shear walls of Appendix A.8 of ACI 318. Adequate provisions are made to transfer wall moments, vertical loads, and shears to the foundation.

The finite element program SLSAP is used to analyze the basemat and the fuel pool walls. Frame analysis is done using computer program STRUDL-II. Concrete beams and columns are designed using the computer programs CBEAM and PCAUC, respectively. The STAND system is used to analyze and design structural steel beams and columns. For design of plate girders, the computer program PLGIRD is used.

Limitation of concrete strain is per ACI 318 for both operating and design-basis loads for all structures, except for structures designed and analyzed for the effects of pipe breaks, including jet impingement, impact, pressurization and flooding outside the containment, where yield line theory (Reference 5) is used. For steel structures, strains are limited to the elastic range under operating and design-basis loadings.

Design and analysis procedures for these structures comply with the portions of ACI-349 Code which are based on ACI 318.

3.8.4.5 Structural Acceptance Criteria

The stresses and strains in the reinforced concrete walls, floor slabs, beams and equipment supports are limited to those specified in ACI 318, except for local stress due to concentrated loads as given in Note f) of Table 3.8-1.2. Serviceability checks are made in accordance with ACI 318 to assure crack control and to keep deflections below the limits prescribed by ACI 318 or to the manufacturers' recommendations for equipment supported by the reinforced concrete. The factors of safety against material strength are contained in the load factors in Table 3.8-1.2 and in the capacity reduction (∅) factors in ACI 318 for the reinforced concrete.

The stresses and strains in the structural steel are limited to those specified in the AISC Specifications, Part 1, when the loading combinations in Table 3.8-2, combinations 1, 2, and 4 through 8, are being designed for.

The appropriate factors of safety against yield are those as discussed in the Commentary to the AISC Specifications. The allowable steel stresses are increased to 1.6 times those specified above, subject to an upper bound of 0.95 f_y (yield stress), when the loading combinations in Table 3.8-2, conditions 9 through 18, are being designed for. In this situation a minimum factor of safety of 1.05 against yield is assured. In either case, deformations of structural steel members are limited because the stresses are kept within the elastic range, and redistribution of loads due to plastic deformations is not permitted except that, consistent with ASME NOG-1, seismic safety features (restraints) that are not necessarily in contact with the runway girder under normal loading conditions may be used as the sole means to resist transverse and vertical seismic reactions for single failure proof cranes qualified in accordance with NUREG-0554. Inelastic deformation of rail components (i.e., rail, rail clips, and rail clip to girder bolts) is permitted provided that the crane is designed to retain control of and hold the load, and the bridge remains on the runway with its wheels prevented from leaving the rails during a seismic event, as required by NUREG-0554 Section 2.5.

The deflections of all critical steel members are calculated and kept below the limits prescribed by the AISC Specifications or manufacturers' recommendation for equipment supported by the steel.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

Noncombustible and fire resistant materials are used wherever necessary throughout the facilities, particularly in areas containing such critical systems as the control room and components of engineered safety features.

The construction materials conform to the standard set forth in Appendix B. Included in this section is a discussion of the quality control procedures employed specifying the frequency and location of sampling, and test requirements for the materials. Cadwelding procedure is also described in detail in this section. See Figures 3.8-36 and 3.8-37 for typical concrete construction details.

3.8.4.7 Testing and Inservice Surveillance Requirements

No preliminary structural integrity or performance tests are conducted. However, rigorous inspection techniques and the quality control procedures described in Appendix B are adopted throughout construction.

Routine periodic inspections of the concrete structures are conducted to check for possible deterioration, excessive cracking, or spalling of the concrete. Similar inspection is made on structural steel members to check for deterioration of surface coatings and abnormal deformations or warpage.

3.8.5 Foundations and Concrete Supports

3.8.5.1 Descriptions of Foundations and Supports

3.8.5.1.1 Foundations

The station is supported by a common reinforced concrete mat. The mat is several feet thick, and is shown on the general arrangement (Section 1.2). The CWSH is supported by a reinforced concrete mat.

The concrete mats bear on the consolidated soil discussed in Section 2.5.

Typical reinforcing patterns at the junctions of the basemat and walls and the basemat and columns are shown in Figure 3.8-37. The mat in the area of the containment, which includes the auxiliary and fuel buildings, is considered part of the containment and is discussed in Subsection 3.8.1.

3.8.5.1.2 Concrete Supports

Seismic Category I equipment is adequately anchored to and/or supported by concrete supports. The concrete supports consist of monolithically poured reinforced concrete pads. The pads are integrally connected to the basemat or floor slabs by dowels. Typical anchor bolt details for Seismic Category I equipment are shown in Figure 3.8-38.

The reactor pedestal which supports the RPV and the reactor shield wall is discussed in Subsection 3.8.3.

3.8.5.2 Applicable Codes, Standards and Specifications

This section lists the codes, specifications, standards of practice, regulatory guides, and other accepted industry guidelines which are adopted to the extent applicable in the design and construction of the foundations and anchorages for Seismic Category I structures and equipment. To eliminate repetition, these codes, standards and specifications are described and discussed in Table 3.8-4 and given a specification reference number. Listed below are the reference numbers for the foundations.

a. 1 through 5, 7, 8, 9;

b. 11 through 14, 16, 17, 18;

c. 23, 24 and 28;

d. 35 through 38;

e. 41, 43, and 46.

3.8.5.3 Load and Loading Combinations

The loads and loading combinations listed and discussed in Subsections 3.8.1.3, 3.8.3.3, and 3.8.4.3 are also applicable to the design of foundations. Refer to Tables 3.8-1.1 and 3.8-1.2 for the load definitions and list of loading combinations that are considered in the design. Stability calculation loads and loading combinations are listed in Table 3.8-1.3.

3.8.5.4 Design and Analysis Procedures

Conventional elastic techniques are used for the design and analysis of all Seismic Category I foundations. Design is based on the ACI 318 Code. All interior and exterior loads on the buildings are transferred to the basemat through elastic deformation of shear walls and columns. The foundation mats are properly sized to accommodate total overturning moments due to wind, tornado or seismic loads without exceeding the allowable soil bearing stress at any point. Horizontal translation due to wind, tornado or seismic loadings is resisted by frictional force between concrete mat and underlying soil. Passive resistance of soil acting against the subgrade walls is neglected. The uplift force due to hydrostatic pressure is deducted from the building dead load to compute the resultant downward load for calculating frictional resistance against sliding. The design and analysis of the foundation complies with the portions of the AC1-349 Code which are based on ACI-318.

In determining the overturning moments due to seismic loads as discussed in Subsection 3.8.5.4 all three components of earthquake are considered acting simultaneous. (Q&R 220.59)

The foundation mats are analyzed as a "mat on elastic foundation" using the finite-element computer program SLSAP-4. The boundary conditions selected for all structural models are determined by evaluating the stiffness (flexural, torsional and axial) of all members connected at a boundary point and represent, to the extent practicable, the actual restraint conditions. Settlements are taken into account by the soil springs modeled for each node point.

The modulus of subgrade reaction is varied within certain limits to determine the effects on critical sections. In general, lower values are used for long term loads and higher values for short term loads.

Structural building supports for rotating or reciprocating (vibratory) Seismic Category I equipment satisfy vibratory design requirements. The equipment foundation design for vibratory equipment satisfies the machine vibration tolerances given in Figure 4 of "Vibration Tolerances" by T. C. Rathbone, Power Plant Engineering, Vol. 43, 1939. This has been accomplished by providing a foundation equipment mass ratio of 2.5 for vibratory equipment which weighs less than 5 kip and by performing appropriate dynamic analyses for vibratory equipment which weighs 5 kip or more.

3.8.5.5 Structural Acceptance Criteria

3.8.5.5.1 Structural Member Design

The acceptance criteria for the reactor containment base slab are as specified in Subsection 3.8.1.5.

The foundations for the main building complex and other Seismic Category I structures are proportioned according to the criteria set forth in Subsection 3.8.4.5.

3.8.5.5.2 Stability

As described in Subsection 3.8.5.4, the basemats are supported on elastic soil springs and overturning is resisted by unequal bearing pressure.

Table 3.8-1.3 lists the loads, load combinations, and factors of safety considered in the foundation stability investigation.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The construction materials for the mat foundations, concrete supports and machinery and equipment anchors conform to the standards set forth in Appendix B. Contained in that appendix is a discussion of the quality control procedures adopted which include the frequency and location of sampling and test requirements for the materials. Cadwelding is described in detail.

3.8.5.7 Testing and Inservice Surveillance Techniques

Routine observations are made of the mat foundations and concrete supports to determine the extent of cracking and settlement. Representative equipment anchor bolts are periodically tested for tightness.

Rigorous inspection during construction in conjunction with the quality control procedures for the structural materials outlined in Appendix B is carried out. Structural integrity and/or performance tests, in addition to those specified herein, are not conducted.

3.8.6 References

1. H. G. Young and L. A. Tate, Design of Liners for Reactor Vessels, Conference on Prestressed Concrete Pressure Vessels, Institute of Civil Engineers, Paper 57, Group J (1967).

2. Nelson Stud Welding Applications in Power Generating Plants, Nelson Stud Welding Company, Lorain, Ohio.

3. J. M. Doyle and S. L. Chu, Liner Plate Buckling and Behavior of Stud and Rib Type Anchors, Proceedings of the First International Conference on Structural Mechanics in Reactor Technology, Vol. 4, Part H, Berlin, Germany (September 1972).

4. C. W. Dunham, "Foundations of Structures," Second Edition, p. 199, McGraw Hill (1962).

5. K. W. Johansen, "Yield-Line Formulae for Slabs," Cement and Concrete Association (1972).

6. ASTM A615-76a, Specifications for Deformed and Plan Billet-Steel Bars for Concrete Reinforcement.

TABLE 3.8-1.1
LOAD COMBINATIONS AND LOAD FACTORS FOR CONTAINMENT STRUCTURES-REINFORCED CONCRETE

LOAD COMBINATION		LOAD CONDITION																												
																								SRV			LOCA			
CATEGORY	NO.	D	L	H	P'	Pa	P_i	P_s	P_o	T_o	T_a	E	E'	W	W'	R_o	R_a	Y_r	Y_m	Y_j	H'	M	F	1V2P	ADS	ALL	MV	PS	CO	CH
I. Construction	1	0.75	0.75	0.75						0.75				0.75																
	2	1.0	1.0	1.0						1.0																				
II. Test	3	1.0	1.0	1.0	1.0					1.0																				
III. Normal/ and and Severe IV. Environmental	4	1.0	1.0	1.0					1.0	1.0		1.0				1.0								1.0		1.0				
	5	1.0	1.0	1.0					1.0	1.0				1.0		1.0								1.0		1.0				
V. Abnormal	6	1.0	1.0	1.0		1.5					1.0						1.0							1.25			1.0	1.0	1.0	1.0
	7	1.0	1.0	1.0			1.5				1.0						1.0							1.25	1.25				1.0	1.0
	8	1.0	1.0	1.0				1.5			1.0						1.0							1.25	1.25					1.0
	9	1.0	1.0	1.0					1.0	1.0						1.0						1.0		1.0		1.0				
VI. Extreme/ Environmental	10	1.0	1.0	1.0					1.0	1.0					1.0	1.0								1.0		1.0				
	11	1.0	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		1.0				
VII. Abnormal/ Severe Environmental	13	1.0	1.0	1.0		1.25					1.0	1.25					1.0	1.0	1.0	1.0				1.0			1.0	1.0	1.0	1.0
	14	1.0	1.0	1.0			1.25				1.0	1.25					1.0	1.0	1.0	1.0				1.0	1.0				1.0	1.0
	15	1.0	1.0	1.0				1.25			1.0	1.25					1.0	1.0	1.0	1.0				1.0	1.0					1.0
	16	1.0	1.0	1.0		1.25					1.0			1.25			1.0	1.0	1.0	1.0				1.0			1.0	1.0	1.0	1.0
	17	1.0	1.0	1.0			1.25				1.0			1.25			1.0	1.0	1.0	1.0				1.0	1.0				1.0	1.0
	18	1.0	1.0	1.0				1.25			1.0			1.25			1.0	1.0	1.0	1.0				1.0	1.0					1.0
VIII. Abnormal/ Extreme Environmental	19	1.0	1.0	1.0		1.0					1.0		1.0				1.0	1.0	1.0	1.0				1.0			1.0	1.0	1.0	1.0
	20	1.0	1.0	1.0			1.0				1.0		1.0				1.0	1.0	1.0	1.0				1.0	1.0				1.0	1.0
	21	1.0	1.0	1.0				1.0			1.0		1.0				1.0	1.0	1.0	1.0				1.0	1.0					1.0
IX. Severe Environ-mental/ Flooded Condition	22	1.0	1.0							1.0				1.0									1.0							
	23	1.0	1.0							1.0		1.0*											1.0							

1-> SERVICE LOADS <-1 ->

FACTORED LOADS

NOTES:

*OBE Flooded Condition

a) For the Construction Category, the wind load for a 10-year recurrence will be used.
b) Loads not applicable to a particular item may be deleted.
c) If for any load combination, the effect of any load other than D reduces the load, it will be deleted from the combination.
d) Each case of SRV Actuation is to be considered one at a time.
e) Each case of LOCA is to be considered one at a time.
f) The 33 1/3% increase in stresses allowed by the ASME B&PV Code, Section III, Division 2, CC-3420 (ACI-ASME 359, 1973) for members subject to wind or earthquake shall not be considered.

TABLE 3.8-1.1 (Cont'd)

Load Categories

I. Construction Category

This category includes all loads during construction.

II. Test Category

This category includes all loads during the structural acceptance test.

III. Normal Category

This category includes all loads on the structure during normal operation and shutdown.

IV. Severe Environmental

This category includes very infrequent loading during the station life, such as operating basis earthquake and design wind.

V. Abnormal Category

This category includes pressure loads and temperature effects from a postulated high-energy pipe break accident within the containment and/or compartment thereof. It includes pipe rupture loads in penetration and impingement loading. It also includes missile effects other than tornado and postulated accident generated missiles.

VI. Extreme Environmental Category

This category includes events which are credible but highly improbable such as a safe shutdown earthquake, and wind forces due to tornado and forces due to tornado generated missiles.

VII. Abnormal/Severe Environmental Category

This extremely unlikely loading is a combination of Categories IV and V.

VIII. Abnormal/Extreme Environmental Category

This extremely unlikely loading is a combination of Categories V and VI.

IX. Severe Environmental/Flooded Category

This category includes very infrequent loading during station life, such as operating basis earthquake, design winds and containment flooding associated loads.

TABLE 3.8-1.1 (Cont'd)

Explanation of Loading Conditions

and Load Categories

D = Dead load of the structure plus any other permanent load; including vertical and lateral pressures of liquids, piping, cable pan and weight of permanent equipment and its normal contents under operating and test conditions.

L = Conventional floor and roof live loads, movable equip-ment loads and other loads which vary in intensity such as lateral soil pressure. Live load intensities may vary from zero to their maximum values to determine the most critical effect upon the structure for the load combination under consideration. To account for the effects of impact, equipment operating support reactions will be increased by the following percentages:

a. elevator supports, 100%;

b. girders and their connections supporting power-operated cranes, 25%;

c. girders and their connections supporting hand-operated cranes, 10%;

d. supports for light machinery, shaft or motor-driven, 20%; and

e. supports for reciprocating machinery or power-driven units, 50%.

H = Hydrostatic pressure load in suppression pool.

Note: Reduced intensities of live loads such as conventional floor loads may be associated with accident and/or severe/extreme environmental conditions.

P_o = Containment normal operating pressure.

P_a* = Containment design accident pressure load due to large size break (DBA).

P_i* = Containment design accident pressure load due to intermediate size break (IBA).

P_s* = Containment design accident pressure load due to small size break (SBA).

P' = Containment test pressure.

R_o = Normal operating or shutdown reactions of piping at supports or anchor points, based on the most critical transient or steady-state condition.

R_a = Pipe reactions under thermal conditions generated by the postulated break.

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* Since these are time-dependent loads, their effect will be superimposed accordingly.

TABLE 3.8-1.1 (Cont'd)

E = Operating basis earthquake (OBE), including dynamic lateral soil pressure and hydrodynamic groundwater pressure.

E' = Safe shutdown earthquake (SSE), including dynamic lateral soil pressure and hydrodynamic groundwater pressure.

F = Loads associated with containment flooding, both hydrostatic and hydrodynamic loads.

T_o = Thermal effects associated with normal operating, shutdown, construction and test conditions; based on the most critical transient or steady-state condition:

a. Climatic temperature ranges

maximum outside temperature - 100° F

minimum outside temperature - 0° F

b. Operating temperature ranges normal operating temperature inside containment - 104° F in general areas, 122° F in some closed compartments, and 95° F in the suppression pool

T_a* = Thermal loads under thermal conditions generated by the postulated break.

H' = Forces associated with the maximum probable flood or seiche. See Section 3.4

W = Design Wind Load See Subsection 3.3.1

W' = Tornado load and loads from tornado generated missiles See Subsections 3.3.2 and 3.5.23

M = Loads associated with missiles other than tornado and postulated accident generated missiles - See Section 3.5

Y_r = Equivalent static load on the structure generated by the reaction on the broken high-energy pipe during the postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load

Y_j = Jet impingement equivalent static load on a structure generated by the postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load

Y_m = Missile impact equivalent static load on a structure Ym generated by or during the postulated break, like pipe whipping, and including an appropriate dynamic factor for the dynamic nature of the load

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* Since these are time-dependent loads, their effect will be superimposed accordingly.

TABLE 3.8-1.1 (Cont'd)

Safety/Relief Valve (SRV) Discharge

SRV IV2P = SRV loading due to subsequent actuation of one safety/relief valve.

SRV/ADS = SRV loading due to seven (ADS) safety/relief valves discharge.

SRV/ALL = SRV loading due to 16 (all) safety/relief valves discharge.

NOTES:

a. The SRV loads are treated as live loads in all load combinations with the exception of the combination that contains $1.5P_a$ where a load factor of 1.25 is applied to the appropriate SRV loads.

b. A single active failure causing one SRV discharge is considered in combination with the design-basis accident (DBA).

c. Appropriate multiple SRV discharge is considered in combination with the small break accident (SBA) and intermediate break accident (IBA).

d. Thermal loads due to SRV discharge are treated as T_o* for normal operation and T_a* for accident conditions.

e. The suppression pool liner is designed in accordance with the ASME Boiler and Pressure Vessel Code, Division 1, Subsection NE, to resist the SRV negative pressure, considering strength, buckling, and low cycle fatigue.

Loss-of-Coolant Accident (LOCA) Loads:

MV = LOCA loading due to main vent clearing

PS = LOCA loading due to pool swell

CO = LOCA loading due to condensation oscillation

CH = LOCA loading due to chugging

* As defined in ACI 359-`1974.

TABLE 3.8-1.2
LOAD COMBINATION AND LOAD FACTORS FOR REINFORCED CONCRETE
(STRUCTURES OTHER THAN CONTAINMENT)

LOADING COMBINATION		LOAD FACTORS																											
			L																			*SRV (h)			*LOCA				
DESCRIPTION	NO.	D	EHL	SLL	C	R_o	P_o	T_o	E	E'	W (a)	Wt	R_a	P_a	T_a	F_a	H'	M	Y_r	Y_j	Y_m	ADS	1V2P	ALL	CH	CO	PS	MV	Design Strength
CONSTRUCTION	1	1.3	1.3		1.3			1.3			1.3																		ACI 318
	2	1.4	1.4		1.7			1.7																					ACI 318
	3	1.1	1.3		1.3	1.3	1.3	1.3																					ACI 318
TEST	4	1.4	1.7		1.7	1.7	1.7	1.7																					ACI 318
NORMAL	5	1.4		1.7	1.7	1.7	1.7	1.7															1.7	1.7					ACI 318
	6	1.4	1.7		1.7	1.7	1.7	1.7			1.7																		ACI 318
SEVERE	7	1.4		1.7	1.7	1.7	1.7	1.7			1.7												1.7	1.7					ACI 318
ENVIRONMENTAL	8	1.2				1.7	1.7	1.7			1.7												1.7	1.7					ACI 318
	9	1.4		1.7	1.7	1.7	1.7	1.7	1.9														1.7	1.7					ACI 318
	10	1.2				1.7	1.7	1.7	1.9														1.7	1.7					ACI 318
	11	1.0		1.0	1.0								1.0	1.5	1.0								1.25		1.0	1.0	1.0	1.0	ACI 318
ABNORMAL	12	1.0		1.0	1.0								1.0	1.5	1.0							1.25			1.0	1.0			ACI 318
	13	1.0		1.0	1.0	1.0	1.0	1.0										1.0					1.0	1.0					ACI 318
	14	1.0		1.0	1.0	1.0	1.0	1.0		1.0													1.0	1.0					ACI 318
EXTREME	15	1.0		1.0	1.0	1.0	1.0	1.0				1.0											1.0	1.0					ACI 318
ENVIRONMENTAL	16	1.0		1.0	1.0	1.0	1.0	1.0									1.0				1.0		1.0	1.0					ACI 318
	17	1.0		1.0	1.0				1.25				1.0	1.25	1.0				1.0	1.0	1.0		1.0		1.0	1.0	1.0	1.0	ACI 318
ABNORMAL/	17A	1.0		1.0	1.0				1.25				1.0			1.0					1.0				1.0	1.0	1.0	1.0	ACI 318
SEVERE	18	1.0		1.0	1.0				1.25				1.0	1.25	1.0				1.0	1.0		1.0	1.0		1.0	1.0			ACI 318
	18A	1.0		1.0	1.0				1.25				1.0			1.0						1.0			1.0	1.0			ACI 318
	19	1.0		1.0	1.0					1.0			1.0	1.0	1.0				1.0	1.0	1.0				1.0	1.0	1.0	1.0	ACI 318
ABNORMAL/	19A	1.0		1.0	1.0					1.0			1.0			1.0							1.0		1.0	1.0	1.0	1.0	ACI 318
EXTREME	20	1.0		1.0	1.0					1.0			1.0	1.0	1.0				1.0	1.0	1.0	1.0			1.0	1.0			ACI 318
	20A	1.0		1.0	1.0					1.0			1.0			1.0						1.0			1.0	1.0			ACI 318

NOTES:

a) For construction combination, wind load for a 10-year recurrence interval shall be used.
b) T_a is based on a temperature corresponding to the pressure, P_a.
c) Loads not applicable to a particular system may be deleted.
d) If for any load combination, the effect of any load other than D reduces the load, it will be deleted from the combination.
e) For E, E', W_t, M & R_a, the resultant effects for both horizontal and vertical force components shall be determined by combining the individual effects by the square root of the sum of the squares.
f) For combinations 13, 17, 18, 19 and 20, local stresses due to concentrated loads Y_r, Y_j, Y_m & M may exceed allowable stresses provided that there will be no loss of function of any safety-related system.
g) For loading combinations 1 through 10, the load factors shown shall be applied using zero values for R_o and T_o. These loads combinations shall also be checked using the values for R_o and T_o, but multiplying the combination by 0.75.
h) SRV and LOCA loads are considered negligible in the radwaste, control, and diesel generator buildings and shall not be used in the analysis and design of these buildings.
* Only one load under each of these loadings shall be considered at one time.

TABLE 3.8-1.2 (Cont'd)

Load Categories

I. Construction Category

This category includes all loads during construction.

II. Test Category

This category includes all loads on the structure during a test of station equipment or systems.

III. Normal Category

This category includes all loads on the structure during normal operation.

IV. Severe Environmental

This category includes very infrequent loadings during the station life such as operating basis earthquake and design wind.

V. Abnormal Category

This category includes pressure loads and temperature effects from a postulated high energy break accident within a building and/or component thereof. It includes pipe rupture loads in penetration and impingement loading. It also includes missile effects other than tornado and postulated accident generated missiles.

VI. Extreme Environmental Category

This category includes events which are credible but highly improbable such as a safe shutdown earthquake, and wind forces due to tornado and forces due to tornado generated missiles.

VII. Abnormal/Severe Environmental Category

This extremely unlikely loading is a combination of Categories IV and V.

VIII. Abnormal/Extreme Environmental Category

This extremely unlikely loading is a combination of Categories V and VI.

Explanation of Loading Conditions
and Load Categories

D = Dead Load of the structure plus any other permanent load; including vertical and lateral pressure of liquids, piping, cable pan and weight of permanent equipment and its normal contents under operating and test conditions.

L = Conventional floor and roof live loads, movable equipment loads (EHL), and other loads which vary in intensity, such as lateral soil pressure. Live load intensities may

TABLE 3.8-1.2 (Cont'd)

vary from zero to their maximum values to determine the most critical effect upon the structure for the load combination under consideration. To account for the effects of impact, equipment operating support reactions will be increased by the following percentages:

a. elevator supports, 100%;

b. girders and their connections supporting power operated cranes, 25%;

c. girders and their connections supporting Hand operated cranes, 10%;

d. supports for light machinery, shaft or motor driven, 20%; and

e. supports for reciprocating machinery or power-driven units, 50%.

The portion of Tables 3.8-1.1, 3.8-1.2 and 3.8-2 addressing "reduced intensities of live load..." does not deviate from the SRP Procedure. Provision of live load "having its full value or being completely absent..." is made in Tables 3.8-1.1 and 3.8-1.2 under Explanation of Loading Conditions and Load Categories. "Live load intensities may vary from zero to their maximum values to determine the most critical effect upon the structure for the load combination under consideration."

The reduced intensities of live loads referred to in the note found in Table 3.8-1.1 are in fact the actual live loads postulated during plant operation. Higher intensities of live load are postulated during plant shutdown to account for dismantled equipment handling and major maintenance operations.

The selection of certain percentages to provide for additional impact live load of specific equipment is in accordance with the general guideline found in Section 1.3.3 of the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings. (Q&R 220.39)

EHL = Movable equipment loads

C = Crane-lifted load, including impact

SLL = Reduced intensities of floor live loads used with seismic loading combinations.

P_a* = Pressure equivalent static load within or across a compartment and/or building, generated by the postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load

R_o = Normal operating or shutdown reactions of piping at supports or anchor points based on the most critical transient or steady-state condition

R_a = Pipe reactions under thermal conditions generated by the postulated break

Y_r = Equivalent static load on the structure generated by the reaction on the broken high-energy pipe during the postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load

TABLE 3.8-1.2 (Cont'd)

Y_j = Jet impingement equivalent static load on a structure generated by the postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load

Y_m = Missile impact equivalent static load on a structure generated by or during the postulated break, like pipe whipping and including an appropriate dynamic factor to account for the dynamic nature of the load

E = Operating basis earthquake (OBE), including dynamic lateral soil pressure and hydrodynamic groundwater pressure

E' = Safe shutdown earthquake (SSE), including dynamic lateral soil pressure and hydrodynamic groundwater pressure

T_o = Thermal effects associated with normal operating, shutdown, construction and test conditions, based on the most critical transient or steady-state conditions

a. Climatic temperature ranges

maximum outside temperature - 100° F

minimum outside temperature - 0° F

b. Operating temperature ranges ambient temperature inside the fuel building, auxiliary building, control building, DG & HVAC radwaste building, and CWSH - 70° F

T_a* = Thermal loads thermal conditions generated by the postulate break

F_a = Flood load generated by a high energy or moderate Energy pipe break outside the containment

H' = Forces associated with the maximum probable flood or seiche (see Section 3.4)

W = Design Wind Load (see Subsection 3.3.1)

W_t = Tornado load and loads from tornado generated missiles Subsections 3.3.2 and 3.5.2.3)

M = Loads associated with missiles other than tornado and postulated accident generated missiles (see Section 3.5)

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* Since these are time-dependent loads, their effect will be superimposted accordingly.

TABLE 3.8-1.2 (Cont'd)

SRV_{IV2P}	=	SRV loading due to one safety/relief valve subsequent actuation
SRV_{ADS}	=	SRV loading due to seven (ADS) safety/relief valves discharge
SRV_{ALL}	=	SRV loading due to 16 (all) safety/relief valves discharge
LOCA MV	=	LOCA loading due to main vent clearing
LOCA PS	=	LOCA loading due to pool swell
LOCA CO	=	LOCA loading due to condensation oscillation
LOCA CH	=	LOCA loading due to chugging

TABLE 3.8-1.3
LOAD COMBINATIONS FOR STRUCTURAL STABILITY OF FOUNDATIONS

LOADING COMBINATION		LOAD FACTORS																SAFETY FACTORS		
											SRV (Notes 3,5)			LOCA (Notes 3,5)						
DESCRIPTION	NO.	D_s	D_e	D_1	L_1	E	E'	W	W_t	H'	IV 2P	ADS	ALL	MVC	PS	CO	CH	OVERTURNING	SLIDING	FLOTATION
SEVERE ENVIRONMENTAL	1	1.0	1.0	1.0	1.0			1.0			1.0		1.0					1.5	1.5	1.5
	2	1.0	1.0	1.0	1.0	1.0					1.0		1.0							
EXTREME ENVIRONMENTAL	3	1.0	1.0	1.0	1.0				1.0		1.0		1.0					1.1	1.1	1.1
	4	1.0	1.0		1.0					1.0	1.0		1.0							
	4a	1.0	1.0	1.0	1.0		1.0				1.0		1.0							
ABNORMAL SEVERE	5	1.0	1.0	1.0	1.0	1.0					1.0			1.0	1.0	1.0	1.0	1.1	1.1	1.1
	5a	1.0	1.0	1.0	1.0	1.0						1.0				1.0	1.0			
ABNORMAL EXTREME	6	1.0	1.0	1.0	1.0		1.0				1.0			1.0	1.0	1.0	1.0	1.1	1.1	1.1
	7	1.0	1.0	1.0	1.0		1.0					1.0				1.0	1.0			

NOTES:

1. D_s = Self Weight of Structure
   D_l = Vertical and Lateral Pressure of Liquid, Groundwater, and Vertical Soil Pressure
   D_e = Actual equipment loads from manufacturers' drawings
   L_l = Lateral soil pressure

2. If for any load combination, the effect of any load other than the dead load reduces the load, it will be deleted from the combination.

3. Only one load under each of these loadings shall be considered at one time.

4. For definition of load combinations and loads not defined in Note 1, refer to Table 3.8-1.2.

5. SRV and LOCA loads are considered negligible in the radwaste, control and diesel generator buildings, and shall not be used in the analysis and design of these buildings.

TABLE 3.8-2

LOAD COMBINATIONS FOR STRUCTURAL STEEL

		LOAD FACTORS	L																			SRV (g) (j)			LOCA (g)(i)(j)				
DESCRIPTION	NO.	D	EHL	SLL	S	C	R_o	P_o	T_o	E	E'	W	W_t	R_a	P_a	T_a	H'	M	Y_r	Y_j	Y_m	ADS	1V2P	ALLV	CH	CO	PS	MVC	ALLOWABLE STRESS
Construction	1	1.0	1.0			1.0			1.0			1.0																	1.33 AISC
	2	1.0	1.0		1.0	1.0			1.0																				AISC
Test	3	1.0	1.0		1.0	1.0	1.0	1.0	1.0																				1.33 AISC
Normal	4	1.0	1.0		1.0	1.0	1.0	1.0	1.0																				AISC
	5	1.0		1.0	1.0	1.0	1.0	1.0	1.0														1.0	1.0					AISC
Severe Environmental	6	1.0		1.0		1.0	1.0	1.0	1.0	1.0													1.0	1.0					AISC
	7	1.0	1.0			1.0	1.0	1.0	1.0			1.0																	AISC
	8	1.0		1.0		1.0	1.0	1.0	1.0			1.0											1.0	1.0					AISC
Abnormal	9	1.0		1.0	1.0	1.0								1.0	1.0	1.0							1.0		1.0	1.0	1.0	1.0	1.6 AISC£ .95 Fy
	10	1.0		1.0	1.0	1.0								1.0	1.0	1.0						1.0			1.0	1.0			1.6 AISC£ .95 Fy
	11	1.0		1.0	1.0	1.0	1.0	1.0	1.0									1.0					1.0	1.0					1.6 AISC£ .95 Fy
Extreme Environmental	12	1.0		1.0		1.0	1.0	1.0	1.0		1.0												1.0	1.0					1.6 AISC£ .95 Fy
	13	1.0		1.0			1.0	1.0	1.0				1.0										1.0	1.0					1.6 AISC£ .95 Fy
	14	1.0		1.0		1.0	1.0	1.0	1.0								1.0						1.0	1.0					1.6 AISC£ .95 Fy
Abnormal/ Severe	15	1.0		1.0		1.0				1.0				1.0	1.0	1.0			1.0	1.0	1.0		1.0		1.0	1.0	1.0	1.0	1.6 AISC£ .95 Fy
	16	1.0		1.0		1.0				1.0				1.0	1.0	1.0			1.0	1.0	1.0	1.0			1.0	1.0			1.6 AISC£ .95 Fy
Abnormal/ Extreme	17	1.0		1.0		1.0					1.0			1.0	1.0	1.0			1.0	1.0	1.0		1.0		1.0	1.0	1.0	1.0	1.6 AISC£ .95 Fy
	18	1.0		1.0		1.0					1.0			1.0	1.0	1.0			1.0	1.0	1.0	1.0			1.0	1.0			1.6 AISC£ .95 Fy

NOTES:

a) For construction combination, wind load for a 10-year recurrence interval shall be used.

b) T_a is based on a temperature corresponding to the pressure, P_a.

c) Loads not applicable to a particular system may be deleted.

d) If for any load combination, the effect of any load other than D reduces the load, it will be deleted from the combination.

e) For E, E', W_t, M & R_a, the resultant effects for both horizontal and vertical force components shall also be determined by combining the individual effects by the square root of the sum of the squares.

f) For loading combinations 2, 4, 5, 6, 7 and 8 use zero values for R_o and T_o with AISC allowables. These combinations shall also be checked using the values of R_o and T_o but the AISC allowables shall be increased by 33%.

g) Only one load under each of these loadings shall be considered at one time.

h) For load categories and load definitions refer to Table 3.8-1.

S - Stability loads. Stability loads are psuedo-static loads applied to a braced steel frame to assure sufficient strength and stiffness for column, beam, and girder stability.

i) The loads due to a pool swell event are applied on the structural steel as pseudo static loads with dynamic load factors consistent with ductility ratios given in Standard Review Plan Section 3.5.3.

j) SRV and LOCA loads are considered negligible in the radwaste, control and diesel generator buildings, and shall not be used in the analysis and design of these buildings.

TABLE 3.8-3
(This table has been deleted)

TABLE 3.8-4
LIST OF SPECIFICATIONS, CODES AND STANDARDS

SPECI-FICATION REFERENCE NUMBER	SPECIFICATION OR STANDARD DESIGNATION	TITLE	EDITION	REMARKS
1	ACI 318-71 or 77 Supplement 1974	Building Code Requirements for Reinforced Concrete	1971 or 1977	Appendix "A" adopted for seismic design
2	ACI 301-72 Revision 1973	Specifications for Structural Concrete for Buildings	1973	
3	ACI 307-68 ANSI A145.1-1968	Recommended Practice for Concrete Formwork	1968	
4	ACI 305-72 ANSI A170.1-1972	Recommended Practice for Hot Weather Concreting	1972	
5	ACI 211.1-74	Recommended Practice for Selecting Proportions for Normal Weight Concrete	1974	
6	ACI 349	Code Requirements for Nuclear Safety Related Structures	1976 and 1980	
7	ACI 315-74	Manual of Standard Practice for Detailing Reinforced Concrete Structures	1974	
8	ACI 306-66	Recommended Practice for Cold Weather Concreting	1966	
9	ACI 309-72	Recommended Practice for Consolidation of Concrete	1972 Title 69-56	
10	(Deleted)			
11	ACI 308-71	Recommended Practice for Curing Concrete	1971 Title 69-1	
12	ACI 212	Guide for Use of Admixtures in Concrete	ACI Journal Sept, 1971 Title 68-56	
13	ACI 214-65 ANSI A146.1-1968	Recommended Practice for Evaluation of Compression Test Results of Field Concrete	1965	
14	ACI 311-64	Recommended Practice for Concrete Inspection	1964	
15	(Deleted)			
16	ACI 304-73	Recommended Practice for Measuring, Mixing, Transporting and Placing Concrete	1973	
17	Report by ACI Committee 304	Placing Concrete by Pumping Methods	ACI Journal May, 1971 Title 68-33	
18	Report by ACI Committee 437 Subcommittee 1	Strength Evaluation of Existing Concrete Structures	ACI Journal Nov, 1967 Title 64-61	
19	(Deleted)			
20	ACI-ASME-359	ASME Boiler & Pressure Vessel Code, Section III, Division 2, Concrete Reactor Vessels and Containments	1973	Issued for trial use and comment

21	AISC-69 or 78	Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings	1969 or 1978	
22	AISI	Specification for the Design of Light Gage Cold-Framed Steel Structural Members	1968 or 1980	
23	AWS D1.1	Structural Welding Code (with required visual inspec tion based upon VWAC, Revision 2)	1976 or 1977	See Note 1
24	AWS D12.1.61	Recommended Practice for Welding Reinforcing Steel, Metal Inserts and Connection in Reinforced Concrete Construction	1961	
25	ASME	ASME Boiler and Pressure Vessel Code, Section III, Division 1, NE	1971 with Summer of 1973 Addenda	For Containment Locks and Hatches
25a	ASME	ASME Boiler and Pressure Vessel Code, Section III, Division 1, NE	1974 with Summer of 1976 Addenda	For Drywell Locks and Hatches
25b	ASME	ASME Boiler and Pressure Vessel Code, Section III, Division 1, NF	1974 with Winter of 1975 Addenda	For Reactor Pedestal
25c	ASME	ASME Boiler and Pressure Vessel Code, Section III, Division 1, ND	1977	For Fuel Pool Gates
25d	ASME	ASME Boiler and Pressure Vessel Code, Section III, Division 2	1977	For Fuel Pool Liners
26	(Deleted)			
27	(Deleted)			
28	ASTM	Annual Books of ASTM Standards		
29	(Deleted)			
30	(Deleted)			
31	UBC	Uniform Building Code	1970 or 1979	
32	(Deleted)			
33	(Deleted)			
34	(Deleted)			
35	NRC Regulatory Guide 1.10	Mechanical Cadweld Splices in Reinforcing Bars of Concrete Containments	Feb. 1, 1971	Withdrawn by the NRC 7/8/81
36	NRC Regulatory Guide 1.12	Instrumentation for Earthquakes	Rev. 1, Apr. 1974	
37	NRC Regulatory Guide 1.13	Spent Fuel Storage Facility Design Basis	Rev. 1, Dec. 1975	
38	NRC Regulatory Guide 1.15	Testing of Reinforcing Bars For Concrete Structures (revision 1)	Dec. 28, 1972	Withdrawn by the NRC 7/8/81
39	NRC Regulatory Guide 1.18	Structural Acceptance Test for Concrete Primary Reactor Containments (Revision 1)	Dec. 28, 1972	Withdrawn by the NRC 7/8/81
40	NRC Regulatory Guide 1.19	Nondestructive Examinations of Primary Containment Liner Welds (Revision 1)	Aug. 11, 1972	Withdrawn by the NRC 7/8/81
41	NRC Regulatory Guide 1.26	Quality Group Classifications and Standards	Rev. 3, Feb. 1976	
42	NRC Regulatory Guide 1.27	Ultimate Heat Sink	Rev. 2, Jan. 1976	
43	NRC Regulatory Guide 1.29	Seismic Design Classification	Rev. 3, Sept. 1978	
44	NRC Regulatory Guide 1.31	Control of Stainless Steel Welding	Rev. 3, Apr. 1976	
45	(Deleted)			

TABLE 3.8-4 (Cont'd)

46	CRSI	Manual of Standard Practice	1973	
47	ANSI N45.2.5	Supplementary QA Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel during Con-struction Phase of Nuclear Power Plants	1974	
48	NRC Regulatory Guide 1.55	Concrete Placement in Category I Structures	Rev. 0, June 1973	Withdrawn by the NRC 7/8/81
49	NRC Regulatory Guide 1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment Systems and Components	June 1973	
50	NRC Regulatory Guide 1.136	Materials for Concrete Contain-ments (Article CC-2000 of the Code for Concrete Reactor Vessels and Containments)	Rev. 1, Oct. 1978	
51	NRC Regulatory Guide 1.142	Safety Related Concrete Structures for Nuclear Power Plants (other than Reactor Vessels and Containments)	Rev. 0 April 1978	

Explanatin of Abbreviations

ACI	-	American Concrete Institute
AISC	-	American Institute of Steel Construction
AISI	-	American Iron and Steel Institute
ANSI	-	American National Standards Institute
API	-	American Petroleum Institute
ASME	-	American Society of Mechanical Engineers
ASTM	-	American Society for Testing and Materials
AWS	-	American Welding Society
CRSI	-	Concrete Reinforcing Steel Institute
NEC	-	National Electric Code
NRC	-	Nuclear Regulatory Commission
UBC	-	Uniform Building Code
VWAC	-	Visual Weld Acceptance Criteria

NOTES:

1. Clarification to and deviation from portions of AWS D1.1 (and VWAC Revision 2 for visual inspection of welds made to the requirements of AWS D1.1) are made based on engineering evaluations.

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS

CONTAINMENT MECHANICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1MC-1	745 ft-0 in.	745 ft-0 in.	225°		18'-0"	Equipment Hatch
1MC-2	741 ft-0 in.	741 ft-0 in.	78°		9'-10"	Personnel Lock
1MC-3	832 ft-3 in.	832 ft-3 in.	60°		9'-10"	Personnel Lock
1MC-4		764 ft-2-7/16 in.	180°	-14 ft-0-1/2 in.	20 in.	Fuel Transfer Tube
1MC-5	770 ft-9-1/6 in.	770 ft-8-11/16 in.	0°	+ 10 ft-6 in.	24 in.	Main Steam "C"
1MC-6	770 ft-9 in.	770 ft-8-5/8 in.	0°	+3 ft-6 in.	24 in.	Main Steam "A"
1MC-7	770 ft-9 in.	770 ft-8-5/8 in.	0°	-3 ft-6 in.	24 in.	Main Steam "D"
1MC-8	770 ft-9-1/16 in.	770 ft-8-11/16 in.	0°	-10 ft-6 in.	24 in.	Main Steam "B"
1MC-9	763 ft-4-1/4 in.	763 ft-3-7/8 in.	0°	+ 7 ft-0 in.	20 in.	Feedwater "A"
1MC-10	763 ft-4-1/4 in.	763 ft-3-7/8 in.	0°	-7 ft-0 in.	20 in.	Feedwater "B"
1MC-11	720 ft-0 in.	720 ft-0 in.	38°		20 in.	RHR Pump Suction "A"
1MC-12	720 ft-0 in.	720 ft-0 in.	323°		20 in.	RHR Pump Suction "B"
1MC-13	720 ft-0 in.	720 ft-0 in.	308°		20 in.	RHR Pump Suction "C"
1MC-14	757 ft-6 in.	757 ft-6 in.	0°		18 in.	RHR Shutdown Suction
1MC-15	764 ft-3 in.	764 ft-3 in.	63°		12 in.	RHR LPCI "A"
1MC-16	748 ft-6 in.	748 ft-6 in.	335°		12 in.	RHR LPCI "B"
1MC-17	748 ft-0 in.	748 ft-0 in.	300°		12 in.	RHR LPCI "C"
1MC-18	740 ft-0 in.	740 ft-0 in.	94°		14 in.	RHR Test to Supp. "A"
1MC-19	740 ft-0 in.	740 ft-0 in.	317°		14 in.	RHR Test to Supp. "C"
1MC-20	752 ft-0 in.	752 ft-0 in.	275°		14 in.	RHR Test to Supp. "B"
1MC-21	739 ft-0 in.	739 ft-0 in.	27°		2 in.	RHR "A" P.R.V.
1MC-22	739 ft-0 in.	739 ft-0 in.	23°			Spare
1MC-23	752 ft-0 in.	752 ft-0 in.	34°		2 in.	RHR"A" P.R.V.
1MC-24	742 ft-0 in.	742 ft-0 in.	23°		12 in.	RHR "A" P.R.V.
1MC-25	740 ft-0 in.	740 ft-0 in.	320°		2 in.	RHR "B" P.R.V. (Pump Suction)
1MC-26	744 ft-9 in.	744 ft-9 in.	339°		12 in.	RHR "B" P.R.V. (Heat Exchanger)
1MC-27	740 ft-0 in.	740 ft-0 in.	334°		1-1/2 in.	RHR "B" P.R.V. (Shutdown Return)
1MC-28	720 ft-0 in.	720 ft-0 in.	355°		6 in.	RCIC Pump Suction
1MC-29	740 ft-0 in.	740 ft-0 in.	303°		1-1/2 in.	RHR "C" P.R.V. (Pump Suction)
1MC-30	740 ft-0 in.	740 ft-0 in.	296°		1-1/2 in.	RHR "C" P.R.V. (Pump Discharge)
1MC-31	745 ft-0 in.	745 ft-0 in.	354°		6 in.	RHR "B" P.R.V. (Crosstie to RCIC)
1MC-32	720 ft-0 in.	720 ft-0 in.	52°		20 in.	LPCS Pump Suction
1MC-33	743 ft-0 in.	743 ft-0 in.	252°		12 in.	HPCS Test to Supp.
1MC-34	720 ft-0 in.	720 ft-0 in.	66°		12 in.	Suppression Pool Clean Up Suction
1MC-35	758 ft-0 in.	758 ft-0 in.	266°		10 in.	HPCS Pump Discharge
1MC-36	752 ft-6 in.	752 ft-6 in.	100°		10 in.	LPCS Pump Discharge
1MC-37	720 ft-0 in.	720 ft-0 in.	243°		20 in.	HPCS Pump Suction
1MC-38	746 ft-0 in.	746 ft-0 in.	97°		4 in.	LPCS P.R.V. (Pump Discharge)
1MC-39	740 ft-0 in.	740 ft-0 in.	357°			Spare
1MC-40	739 ft-0 in.	739 ft-0 in.	30°		2 in.	RCIC Min. Flow
1MC-41	739 ft-11-5/8 in.	740 ft-0 in.	9°		12 in.	RCIC Turbine Steam EXH

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

CONTAINMENT MECHANICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1MC-42	756 ft-4-3/4 in.	756 ft-4-3/4 in.	0°	-11 ft-0 in.	4 in.	RCIC Head Spray
1MC-43	758 ft-0-3/8 in.	757 ft-10-1/8 in.	0°	-5 ft-0 in.	8 in.	RCIC Turbine Steam Supply
1MC-44	745 ft-0 in.	745 ft-0 in.	21°		3 in.	RCIC Turbine Vacuum Breaker
1MC-45	756 ft-6-3/8 in.	756 ft-6 in.	0°	-15 ft-0 in.	3 in.	Main Steam Drain
1MC-46	756 ft-9 in.	756 ft-9 in.	95°		10 in.	Component Cooling Water Supp.
1MC-47	756 ft-9 in.	756 ft-9 in.	98°		10 in.	Component Cooling Water Return
1MC-48	758 ft-0 in.	758 ft-0 in.	121°		3 in.	Shutdown Service Water Supply
1MC-49	763 ft-6 in.	763 ft-6 in.	124°		1 in.	Breathing Air
1MC-50	790 ft-0 in.	790 ft-0 in.	95°		4 in	Make-up Condensate Supply
1MC-51	790 ft-0 in.	790 ft-0 in.	98°			Spare
1MC-52	758 ft-0 in.	758 ft-0 in.	127°		8 in.	Fuel Pool Cooling & Cleanup
1MC-53	758 ft-0 in.	758 ft-0 in.	145°		10 in.	Fuel Pool Cooling & Cleanup
1MC-54	762 ft-3 in.	762 ft-3 in.	0°	+ 2 ft-3 in.		Spare
1MC-55	743 ft-0 in.	743 ft-0 in.	266°			Spare
1MC-56	753 ft-0 in.	753 ft-0 in.	312°		10 in.	F.P. Containment Standpipe
1MC-57	769 ft-11 in.	769 ft-11 in.	65°		3 in.	Instrument Air
1MC-58	770 ft-0 in.	770 ft-0 in.	61°		1 in.	Instrument Booster Air
1MC-59	770 ft-0 in.	770 ft-0 in.	68°		3 in.	Service Air
1MC-60	756 ft-6 in.	756 ft-6 in.	0°	+ 11 ft-3 in.	6 in.	RWCU Pump Supply
1MC-61	758 ft-3 in.	758 ft-3 in.	0°	+ 7 ft-11 in.	4 in.	RWCU Pump Return
1MC-62	763 ft-6 in.	763 ft-6 in.	294°		2 in.	Hydrogen Recombiner to Containment
1MC-63	758 ft-0 in.	758 ft-0 in.	187°		2 in.	C.R.D. Pump Discharge
1MC-64	763 ft-8-7/8 in.	763 ft-8-7/8 in.	0°	+ 15 ft-0 in.	4 in.	RWCU to RHR Return
1MC-65	763 ft-8-7/8 in.	763 ft-8-7/8 in.	0°	-15 ft-0 in.	2 in.	Radwaste Reprocessing & Disposal
1MC-66	752 ft-0 in.	752 ft-0 in.	75°			Spare
1MC-67	743 ft-0 in.	743 ft-0 in.	245°		6 in.	Containment Service Air (Cnmt. Press)
1MC-68	752 ft-0 in.	752 ft-0 in.	82°		½ & ¾ in.	Process Sampling
1MC-69	758 ft-0 in.	758 ft-0 in.	124°		3 in.	Containment Equipment Drains
1MC-70	763 ft-9 in.	763 ft-9 in.	277°		3 in.	Containment Floor Drains
1MC-71	752 ft-0 in.	752 ft-0 in.	58°		2 in.	Hydrogen Recombiner from Contnt.
1MC-72	752 ft-0 in.	752 ft-0 in.	66°		2 in.	Hydrogen Recombiner to Contnt.
1MC-73	745 ft-0 in.	745 ft-0 in.	195°			Spare
1MC-74	762 ft-3 in.	762 ft-3 in.	0°	-11 ft-9 in.	6 in.	RT Decontamination
1MC-75	764 ft-0 in.	764 ft-0 in.	192°			Spare
1MC-76	740 ft-0 in.	740 ft-0 in.	323°		1-1/4 in.	RHR P.R.V. (Drain)
1MC-77	739 ft-11-5/8 in.	740 ft-0 in.	4°			Spare
1MC-78	751 ft-0 in.	751 ft-0 in.	281°		4 in.	Component Cooling Water Supply
1MC-79	748 ft-0 in.	748 ft-0 in.	259°		10 in.	Suppression Pool Clean-Up Return
1MC-80	753 ft-0 in.	753 ft-0 in.	40°			Spare
1MC-81	742 ft-0 in.	742 ft-0 in.	27°		6 in.	Fire Protection
1MC-82	758 ft-0 in.	758 ft-0 in.	154°		10 in.	Fire Protection

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

CONTAINMENT MECHANICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1MC-83	768ft-0 in.	768ft-0 in.	127°			Spare
1MC-84	751ft-9 in.	751ft-9 in.	44°			Spare
1MC-85	753ft-0 in.	753ft-0 in.	307°		6 in.	Cycle Condensate
1MC-86	762ft-3 in.	762ft-3 in.	0°	+ 12ft-0 in.	4 in.	RWCU to Condenser
1MC-87	753ft-0 in.	753ft-0 in.	48°		1½ in.	RHR "A" P.R.V.
1MC-88	743ft-0 in.	743ft-0 in.	335°		4 in.	Component Cooling Water Return
1MC-89	765ft- 10-7/16 in.	765ft- 10-7/16 in.	25°		1½ in.	RHR Ht. Exch. Shell Vent
1MC-101	790ft-0 in.	790ft-0 in.	146°		36 in.	Cont. Vent Air Supply
1MC-102	772ft-6 in.	772ft-6 in.	146°		36 in.	Cont. Vent Air Purge & Exhaust
1MC-103	768ft-0 in.	768ft-0 in.	151°		10 in.	Cont. Cooling Unit Chilled Water Supp.
1MC-104	768ft-0 in.	768ft-0 in.	145°		10 in.	Cont. Cooling Unit Chilled Water Return
1MC-105	788ft-0 in.	788ft-0 in.	102°			Spare
1MC-106	782ft-6 in.	782ft-6 in.	113°		12 in.	Continuous Cnmt. Purge Air Exh.
1MC-107	769ft- 10-1/2 in.	769ft- 10-1/2 in.	205°		10 in.	Drywell Cooling Unit Chilled Water Supp.
1MC-108	769ft- 10 in.	769ft- 10 in.	202°		10 in.	Drywell Cooling Unit Chiller Water Return
1MC-109	769ft- 10-1/4 in.	769ft- 10-1/4 in.	199°		10 in.	Drywell Cooling Unit Chilled Water Supp.
1MC-110	769ft- 9-3/4 in.	769ft- 9-3/4 in.	196°		10 in.	Drywell Cooling Unit Chiller Water Return
1MC-111	792ft-6 in.	792ft-6 in.	102°			Spare
1MC-112	792ft-0 in.	792ft-0 in.	113°			Spare
1MC-113	782ft-6 in.	782ft-6 in.	117°		12 in.	Continuous Cnmt. Purge Air Supply
1MC-114	782ft-6 in.	782ft-6 in.	51°			Spare
1MC-115	782ft-6 in.	782ft-6 in.	102°			Spare
1MC-116	782ft-6 in.	782ft-6 in.	93°		2 in.	Standby Liquid Control
1MC-150	782ft-6 in.	782ft-9 in.	62°		3/4 in.	Cnmt. Pressure Monitors
1MC-151	782ft-6 in.	782ft-9 in.	65°		3/4 in.	Cnmt. Pressure Monitors/ DW Pressure Recirculation
1MC-152	767ft-0 in.	767ft-3 in.	182°		3/4 in.	Containment Monitoring
1MC-153	782ft-6 in.	782ft-9 in.	107°		3/4 in.	Drywell Pressure
1MC-154	770ft-0 in.	770ft-3 in.	298°			Spare
1MC-155	790ft-0 in.	790ft-3 in.	68°			Spare
1MC-156	792ft-0 in.	792ft-3 in.	235°		3/4 in.	Containment Pressure (SGTS Train A)/ DW Air Temp/Humidity Inst. Panel

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

CONTAINMENT MECHANICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1MC-157	792ft-0 in.	792ft-3 in.	248°		¾ in.	Suppression Pool Water Level Monitoring
1MC-158	782ft-6 in.	782ft-3 in.	60°			Spare
1MC-159	763ft-6 in.	763ft-3 in.	120°			Spare
1MC-160	767ft-0 in.	766ft-9 in.	190°		¾ in.	Containment Monitoring System
1MC-161	770ft-0 in.	769ft-9 in.	294°			Spare
1MC-162	790ft-0 in.	790ft-3 in.	205°			Spare
1MC-163	790ft-0 in.	790ft-3 in.	210°			Spare
1MC-164	765ft-0 in.	764ft-9 in.	150°		¾ in.	Suppression Pool Makeup
1MC-165	792ft-6 in.	792ft-3 in.	128°		¾ in.	Containment Differential Pressure
1MC-166	789ft-0 in.	788ft-9 in.	291°		2 in.	Hydrogen Recombiner from Containment
1MC-167	788ft-0 in.	788ft-3 in.	117°		¾ in.	Containment Pressure (SGTS Train B)
1MC-168	789ft-0 in.	789ft-3 in.	280°		¾ in.	Containment Differential Pressure
1MC-169	793ft-0 in.	792ft-9 in.	117°		¾ in.	Continuous Containment Purge Damper Control
1MC-170	790ft-0 in.	790ft-3 in.	65°			Spare
1MC-171	767ft-0 in.	767ft-3 in.	185°		¾ in.	Suppression Pool Makeup
1MC-172	773ft-1-1/8 in.	773ft-4-15/16 in.	300°		1½ in.	HR Ht. Exch. Shell Vent
1MC-173	788ft-0 in.	788ft-3 in.	242°	-3½ in	¾ in.	Containment Monitoring System
1MC-174	788ft-0 in.	788ft-3 in.	128°			Spare
1MC-175	788ft-0 in.	788ft-3 in.	315°			Spare
1MC-176	788ft-0 in.	788ft-3 in.	62°			Spare
1MC-177	720ft-0 in.	720ft-0 in.	60°		1¼ in.	Supp. Pool Water Level (RCIC)
1MC-178	745ft-0 in.	745ft-0 in.	55°			Spare
1MC-179	720ft-0 in.	720ft-0 in.	150°		1¼ & ¾ in.	H.P. Core Spray System & Suppression Pool Make-up
1MC-180	745ft-0 in.	745ft-0 in.	150°		1¼ in.	H.P. Core Spray System
1MC-181	720ft-0 in.	720ft-0 in.	203°		1¼ in.	Supp. Pool Water Level
1MC-182	742ft-6 in.	742ft-6 in.	203°			Spare
1MC-183	720ft-0 in.	720ft-0 in.	260°		1¼ in.	Supp. Pool Water Level
1MC-184	742ft-6 in.	742ft-6 in.	260°			Spare
1MC-200	769ft-6 in.	769ft-6 in.	76°		¾ in.	Supp. Pool Water Level (RCIC)
1MC-201	769ft-6 in.	769ft-6 in.	79°			Spare
1MC-202	769ft-6 in.	769ft-6 in.	82°			Spare
1MC-203	782ft-6 in.	782ft-6 in.	298°		¾ in.	Containment Monitoring
1MC-204	782ft-6 in.	782ft-6 in.	295°		3 in.	S/D Service Water
1MC-205	782ft-6 in.	782ft-6 in.	291°		3 in.	S/D Service Water
1MC-206	788ft-0 in.	788ft-0 in.	245°		1 in.	Instrument Air
1MC-207	745ft-0 in.	745ft-0 in.	190°			Spare
1MC-208	768ft-0 in.	768ft-0 in.	120°		3 in.	S/D Service Water Return
1MC-209	745ft-0 in.	745ft-0 in.	185°			Spare
1MC-210	759ft-6 in.	759ft-6 in.	0°	+4ft-6 in.	¾ in.	Post Accident Sample
1MC-211	756ft-6 in.	756ft-6in.	0°	+4ft-6 in.		Spare

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

CONTAINMENT ELECTRICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1P1B-1	773ft-0 in.	773ft-0 in.	42°30'		18 in.	Recirc. Pump 1A (P)
1P2B-1	773ft-0 in.	773ft-0 in.	317°30'		18 in.	Recirc. Pump 1B (P)
1K1E-2	769ft-0 in.	769ft-0 in.	37°30'		12 in.	Instrumentation 1 (K)
1K2B-1	769ft-0 in.	769ft-0 in.	302°30'		12 in.	Instrumentation (K)
1K1B-2	769ft-0 in.	769ft-0 in.	57°30'		12 in.	Instrumentation (K)
1K2B-2	769ft-0 in.	769ft-0 in.	322°30'		12 in.	Instrumentation (K)
1K1N	792ft-0 in.	792ft-0 in.	42°30'		12 in.	Neutron Monitoring System 1 (K)
1K2E-1	792ft-0 in.	792ft-0 in.	307°30'		12 in.	Instrumentation 2 (K)
1K1E	792ft-0 in.	792ft-0 in.	52°30'		12 in.	Instrumentation 1 (K)
1K4N	792ft-0 in.	792ft-0 in.	315°		12 in.	Neutron Monitoring System 4 (K)
1C1E	794ft-0 in.	794ft-0 in.	40° 0'		12 in.	Reactor Protection System 1 (C)
1C2E-1	794ft-0 in.	794ft-0 in.	305°		12 in.	Reactor Protection System 2 (C)
1C3E	794ft-0 in.	794ft-0 in.	55° 0'		12 in.	Reactor Protection System 3 (C)
1C4E	794ft-0 in.	794ft-0 in.	317°30'		12 in.	Reactor Protection System 4 (C)
1C1B-2	771ft-0 in.	771ft-0 in.	40° 0'		12 in.	Information (C)
1C2B-1	771ft-0 in.	771ft-0 in.	310°		12 in.	Information (C)
1C1B-1	771ft-0 in.	771ft-0 in.	50°		12 in.	Information (C)
1C2B-2	771ft-0 in.	771ft-0 in.	320° 0'		12 in.	Information (C)
1K1B-1	769ft-0 in.	769ft-0 in.	52°30'		12 in.	Instrumentation (K)
1K2B-3	769ft-0 in.	769ft-0 in.	307°30'		12 in.	Instrumentation (K)
1K2N	794ft-6 in.	794ft-6 in.	240°		8 in.	Neutron Monitoring System 2 (K)
1K3N	794ft-6 in.	794ft-6 in.	140°		8 in.	Neutron Monitoring System 3 (K)
1P1E-1	796ft-0 in.	796ft-0 in.	42°30'		12 in.	Engineered Safety Feature 1 (P)
1P2E-1	796ft-0 in.	796ft-0 in.	307°30'		12 in.	Engineered Safety Feature 2 (P)
1P2E-3	796ft-0 in.	796ft-0 in.	52°30'		12 in.	Engineered Safety Feature 2 (P)
1P1B-4	773ft-0 in.	773ft-0 in.	47°30'		12 in.	Balance of Plant (P)
1P2B-4	773ft-0 in.	773ft-0 in.	312°30'		12 in.	Balance of Plant (P)
1SP-1	794ft-0 in.	794ft-0 in.	45°			Spare
1C2E-2	794ft-0 in.	794ft-0 in.	310°		12 in.	Engineered Safety Feature 2 (C)
1P1E-2	794ft-0 in.	794ft-0 in.	50°		12 in.	Engineered Safety Feature 1 (P)
1K1E-1	792ft-0 in.	792ft-0 in.	37°30'		12 in.	Control Rod Indication 1 (K)
1K2E-2	792ft-0 in.	792ft-0 in.	302°30'		12 in.	Control Rod Indication 2 (K)
1K3E	792ft-0 in.	792ft-0 in.	57°30'		12 in.	Instrumentation 3 (K)
1K4E	792ft-0 in.	792ft-0 in.	320°		12 in.	Instrumentation 4 (K)
1P1B-3	773ft-0 in.	773ft-0 in.	52°30'		12 in.	Balance of Plant (P)
1P2B-3	773ft-0 in.	773ft-0 in.	307°30'		12 in.	Balance of Plant (P)
1C1B-3	771ft-0 in.	771ft-0 in.	55°		12 in.	Information (C)
1C2B-3	771ft-0 in.	771ft-0 in.	305°		12 in.	Information (C)
1P2E-2	796ft-0 in.	796ft-0 in.	302°30'		12 in.	Engineered Safety Feature 2 (P)
1P1B-2	773ft-0 in.	773ft-0 in.	57°30'		12 in.	Balance of Plant (P)
1P2B-2	773ft-0 in.	773ft-0 in.	302°30'		12 in.	Balance of Plant (P)

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

CONTAINMENT ELECTRICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1P1B-5	771ft-0 in.	771ft-0 in.	35°		12 in.	Balance of Plant (P)
1SP-3	771ft-0 in.	771ft-0 in.	325°			Spare
1K1B-3	775ft-9 in.	775ft-9 in.	192°		12 in.	Testing Instrumentation (K)
1K1B-4	775ft-9 in.	775ft-9 in.	205°		12 in.	Testing Instrumentation (K)

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

DRYWELL MECHANICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1MD-1	744ft-0 in.	744ft-0 in.	225°		16ft-0 in.	Equipment Hatch
1MD-2	741ft-0 in.	741ft-0 in.	90°		10ft-0 in.	Personnel Lock
1MD-3			90°		24 in.	Manhole
1MD-4	786ft-0 in.	786ft-0 in.	60°	+0ft-8 in.	3 in.	Standby Liquid Conrol
1MD-5	771ft-0-5/8 in.	771ft-0 in.	0°	+10ft-6 in.	24 in.	Main Steam "C"
1MD-6	771ft-0-1/2 in.	770ft-11-7/8 in.	0°	+3ft-6 in.	24 in.	Main Steam "A"
1MD-7	771ft-0-1/2 in.	770ft-11-7/8 in	0°	-3ft-6 in.	24 in.	Main Steam "D"
1MD-8	771ft-0-5/8 in.	771ft-0 in.	0°	-10ft-6 in.	24 in.	Main Steam "B"
1MD-9	763ft-7-1/8 in.	763ft-6-1/2 in.	0°	+7ft-0 in.	20 in.	Feedwater "A"
1MD-10	763ft-7-1/8 in.	763ft-6-1/2 in.	0°	-7ft-0 in.	20 in.	Feedwater "B"
1MD-11	744ft-0 in.	744ft-0 in.	145°		3/4 in.	RR Pump Seal Purge "A"
1MD-12	744ft-0 in.	744ft-0 in.	325°		3/4 in.	RR Pump Seal Purge "B"
1MD-13	763ft-0 in.	763ft-0 in.	119°		3/4 in.	RR Process Sampling
1MD-14	757ft-6 in.	757ft-6 in.	0°		18 in.	RHR Shutdown Suction
1MD-15	761ft-3-1/2 in.	761ft-3-1/2 in.	45°		12 in.	RHR/LPCI "A"
1MD-16	764ft-0-1/2 in.	764ft-0-1/2 in.	210°		12 in.	RHR/LPCI "B"
1MD-17	764ft-0-1/2 in.	764ft-0-1/2 in.	160°		12 in.	RHR/LPCI "C"
1MD-18	731ft-11-5/8 in.	762ft-11 in.	16°		10 in.	Main Steam RV Vents
1MD-19	731ft-11-9/16 in.	726ft-11-5/16 in.	37°		10 in.	Main Steam RV Vents
1MD-20	731ft-11-3/4 in.	726ft-11-9/16 in.	58°		10 in.	Main Steam RV Vents
1MD-21	731ft-10-7/8 in.	726ft-10-1/4 in.	79°		10 in.	Main Steam RV Vents
1MD-22	731ft-10-7/8 in.	726ft-10-3/16 in.	101°		10 in.	Main Steam RV Vents
1MD-23	731ft-11-3/16 in.	726ft-10-1/8 in.	122°		10 in.	Main Steam RV Vents
1MD-24	731ft-11-5/16 in.	726ft-10-13/16 in.	143°		10 in.	Main Steam RV Vents
1MD-25	731ft-11-1/2 in	726ft-10-1/2 in.	164°		10 in.	Main Steam RV Vents
1MD-26	732ft-3/16 in.	726ft-11-3/8 in.	185°		10 in.	Main Steam RV Vents
1MD-27	731ft-11-1/4 in.	726ft-10-7/16 in.	206°		10 in.	Main Steam RV Vents
1MD-28	731ft-10-5/8 in.	726ft-10 in.	238°		10 in.	Main Steam RV Vents
1MD-29	731ft-11-1/16 in.	726ft-10-13/16 in.	259°		10 in.	Main Steam RV Vents
1MD-30	731ft-11¼ in.	726ft-10-5/8 in.	281°		10 in.	Main Steam RV Vents
1MD-31	731ft-10-15/16 in.	726ft-10-9/16 in.	302°		10 in.	Main Steam RV Vents
1MD-32	731ft-10-7/16 in.	726ft-10-15/16 in.	323°		10 in.	Main Steam RV Vents
1MD-33	731ft-9-3/4 in.	726ft-10-3/4 in.	344°		10 in.	Main Steam RV Vents
1MD-34				29' 8-5/8"*		Drywell Head
1MD-35	769ft-5¼ in.	765ft-5¼ in.	270°		10 in.	HPCS Pump Discharge
1MD-36	769ft-5¼ in.	769ft-5¼ in.	90°		10 in.	LPCS Pump Discharge
1MD-37	764ft-9-9/16 in.	769ft-9-1/16 in.	76°24'	7'-5" x 5'-0"**		CRD Insert/Withdrawal Quad 1
1MD-38	764ft-9-11/16 in.	769ft-9-3/16 in.	101°22'	5'-9" x 5'-0"**		CRD Insert/Withdrawal Quad 3
1MD-39	764ft-9-9/16 in.	769ft-9-1/16 in.	256°24'	7'-5" x 5'-0"**		CRD Insert/Withdrawal Quad 2
1MD-40	764ft-9-11/6 in.	769ft-9-3/16 in.	281°22'	5'-9" x 5'-0"**		CRD Insert/Withdrawal Quad 4
1MD-41	748 ft-0 in.	748 ft-0 in.	298°		12 in	Spare (Capped)

* Inside diameter of drywell head.
**Dimentions of insert/withdrawal assemblies.

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

DRYWELL MECHANICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1MD-42	756ft-4-3/4 in.	756ft-4-3/4 in.	0°	-11ft-0 in.	4 in.	RCIC Head Spray
1MD-43	758ft-1-1/16 in.	758ft-9/16 in.	0°	-4ft-9-7/16 in.	8 in.	Steam to RHR/RCIC
1MD-44	764ft-0-1/2 in.	764ft-0-1/2 in.	204°		12 in.	Structural Integrity Test Cable
1MD-45	756ft-6 in.	756ft-6 in.	0°	-15ft-0 in.	3 in.	Main Steam Drain
1MD-46	744ft-3 in.	744ft-3 in.	54°		6 in.	Component Cooling Water Supply
1MD-47	745ft-9 in.	745ft-9 in.	67°		6 in.	Component Cooling Water Return
1MD-48	748ft-0 in.	748ft-0 in.	72°		¾ in.	Nuclear Boiler Instrumentation
1MD-49	747ft-0 in.	747ft-0 in.	54°		4 in.	Spare (Capped)
1MD-50	790ft-0 in.	790ft-0 in.	70°		12 in.	Spare (Capped)
1MD-51	792ft-8 in.	792ft-8 in.	260°		12 in.	For Electrical Use
1MD-52						Not Used
1MD-53	775ft-9 in.	775ft-9 in.	121°		4 in.	Drywell Cooling
1MD-54	790ft-0 in.	790ft-0 in.	90°		3 in.	D. C. Welding Recpt.
1MD-55	764ft-0-1/2 in.	764ft-0-1/2 in.	154°		½ & 1 in.	Recirc. Pump 1A (Hyd. Line to Pwr. Unit)
1MD-56	748ft-0 in.	748ft-0 in.	285°		10 in.	Spare (Capped)
1MD-57	764ft-0 in.	764ft-0 in.	45°		3 in.	Instrument Air
1MD-58	759ft-0 in.	759ft-0 in.	52°		1 in.	Instrument Booster Air
1MD-59	761ft-6 in.	761ft-6 in.	52°		3 in.	Service Air
1MD-60	756ft-6 in.	756ft-6 in.	0°	+11ft-3 in.	6 in.	Reactor Water Clean-up Pump Suct.
1MD-61	748ft-0 in.	748ft-0 in.	170°		10 in.	Spare (Capped)
1MD-62	748ft-0 in.	748ft-0 in.	330°		6 in.	Rx Recirculation Pump Motor Feed
1MD-63	764ft-0 in.	764ft-0 in.	190°		2 in.	Spare (Capped)
1MD-64	790ft-0 in.	790ft-0 in.	138°		18 in.	High Range Radiation Monitor
1MD-65	748ft-0 in.	748ft-0 in.	177°		18 in.	Spare (Capped)
1MD-66	748ft-0 in.	748ft-0 in.	150°		18 in.	Spare (Capped)
1MD-67	758ft-0 in.	758ft-0 in.	182°		6 in.	Spare (Capped)
1MD-68	748ft-0 in.	748ft-0 in.	160°		10 in.	Spare (Capped)
1MD-69	752ft-6 in.	752ft-6 in.	167°		3 in.	Drywell Equipment Drain
1MD-70	750ft-0 in.	750ft-0 in.	299°		3 in.	Drywell Floor Drain
1MD-71	762ft-3 in.	762ft-3 in.	0°	+2ft-3 in.	3 in.	Spare (Capped)
1MD-72	790ft-0 in.	790ft-0 in.	130°		10 in.	Drywell Vacuum Breaker
1MD-73	792ft-8 in.	792ft-8 in.	270°		8 in.	Spare (Capped)
1MD-74						Not used
1MD-75						Not used
1MD-76	761ft-0 in.	761ft-0 in.	219°		2 in.	Spare (Capped)
1MD-77						Not used
1MD-78	784 ft-8 in.	784ft-8 in.	309°		¾ in.	Reactor Pressure Vessel Level “D”
1MD-79	723ft-11 in.	723ft-11 in.	69°		6 in.	Hydrogen Bubbler Pipe
1MD-80	723ft-11 in.	723ft-11 in.	249°		6 in.	Hydrogen Bubbler Pipe
1MD-81	743ft-0 in.	743ft-8 in.	35°	-5ft-0 in.	3/8 in.	TIP System A
1MD-82	739ft-1 in.	739ft-9 in.	35°	+5ft-0 in.	3/8 in.	TIP System B

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

DRYWELL MECHANICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1MD-83	743 ft-0-1/8 in.	743 ft-8 in.	35°		3/8 in.	TIP System C
1MD-84	739 ft-1-1/8 in.	739 ft-9 in.	35°		3/8 in.	TIP System D
1MD-85						Not Used
1MD-86	739 ft-9 in.	739 ft-9 in.	35°	+2 ft-6 in.	½ in.	TIP System Purge Supply
1MD-87	776 ft-0 in.	776 ft-0 in.	44°		8 in.	Spare (Capped)
1MD-88	761 ft-0 in.	761 ft-0 in.	230°		8 in.	Spare (Capped)
1MD-89	762 ft-3 in.	762 ft-3 in.	0°	-11 ft-9 in.	3/8 & ½ in.	Post Accident Sample
1MD-90	790 ft-0 in.	790 ft-0 in.	80°		6 in.	Spare (Capped)
1MD-91	790 ft-0 in.	790 ft-0 in.	120°		6 in.	Spare (Capped)
1MD-92	792 ft-8 in.	792 ft-8 in.	280°		6 in.	Spare (Capped)
1MD-93	784 ft-8 in.	784 ft-8 in.	320°30'			Spare (Capped)
1MD-94	764 ft-9 in.	764 ft-9 in.	314°		½ to 1 ½ in.	Recirc. Pump 1B – RHR
1MD-95	748 ft-0 in.	748 ft-0 in.	350°		2 in.	Spare (Capped)
1MD-96	792 ft-8 in.	792 ft-8 in.	242°		1 in.	Instrument Air
1MD-97	776 ft-0 in.	776 ft-0 in.	190°		4 in.	Spare (Capped)
1MD-98						Not Used
1MD-99	748 ft-0 in.	748 ft-0 in.	0°		4 in.	Spare (Capped)
1MD-100	749 ft-0 in.	749 ft-0 in.	265°		6 in.	Drywell Pressure Test
1MD-101	793 ft-9-1/2 in.	793 ft-9-1/2 in.	313°	-5/8 in.	24 in.	Drywell Purge Air Inlet
1MD-102	776 ft-11 in.	776 ft-11 in.	135°	+8 ft-0 in.	24 in.	Drywell Purge Air Outlet
1MD-103	776 ft-11 in.	776 ft-11 in.	135°	+14 ft-8 in.	2 in.	Spare (Capped)
1MD-104	791 ft-3 in.	791 ft-3 in.	304°30'		24 in.	High Range Radiation Detector
1MD-105	792 ft-0 in.	792 ft-0 in.	113°		24 in.	Spare (Capped)
1MD-106	792 ft-0 in.	792 ft-0 in.	98°		1 in.	Breathing Air
1MD-107	770 ft-0 in.	770 ft-0 in.	219°	+11 ft-6 in.	10 in.	Drywell Chilled Water Supply
1MD-108	770 ft-0 in.	770 ft-0 in.	219°	+8 ft-6 in.	10 in.	Drywell Chilled Water Return
1MD-109	771 ft-3 in.	771 ft-3 in.	200°		10 in.	Drywell Chilled Water Supply
1MD-110	771 ft-3 in.	771 ft-3 in.	194°		10 in.	Drywell Chilled Water Return
1MD-111	Located on	Top of Drywell			6 in.	From Drywell to H2 Mix. Fan "A"
1MD-112	Located on	Top of Drywell			6 in.	To Drywell from H2 Mix Fan "A"
1MD-113	785 ft-0 in.	785 ft-0 in.	218°	+0 ft-2-1/2 in.	6 in.	From Drywell to H2 Mix. Fan "B"
1MD-114	785 ft-0 in.	785 ft-0 in.	225°	+0 ft-2-5/16 in.	6 in.	To drywell from H2 Mix. Fan "B"
1MD-115	787 ft-0 in.	787 ft-0 in.	120°		4 in.	Spare (Capped)l
1MD-116	785 ft-0 in.	785 ft-0 in.	315°			Spare (Capped)
1MD-117	790 ft-0 in.	790 ft-0 in.	45°		10 in.	Drywell Vacuum Breaker
1MD-118	790 ft-0 in.	790 ft-0 in.	105°			Spare (Capped)
1MD-119	764 ft-3 in.	764 ft-3 in.	215°		10 in.	Drywell Vacuum Breaker
1MD-120	764 ft-0 in.	764 ft-0 in.	309°		10 in.	Drywell Vacuum Breaker
1MD-121						Not Used
1MD-122						Not Used

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

DRYWELL MECHANICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1MD-123	764 ft-0 in.	764 ft-0 in.	170°		4 in.	Spare (Capped)
1MD-124	764 ft-0 in.	764 ft-0 in.	180°		4 in.	Fire Protection
1MD-125	766 ft-0 in.	766 ft-0 in.	47°		3 & 1-1/2 in.	Cycled Condensate & RHR
1MD-126	763 ft-3 in.	763 ft-3 in.	270°	+2 ft-0 in.	4 in.	Comp. Cooling Water (Supply)
1MD-127	770 ft-0 in.	770 ft-0 in.	322°		4 in.	Comp. Cooling Water (Return)
1MD-128	791'-3"	790'-10"	50°		¾ in.	Reactor Pressure Level "A"
1MD-129	791'-3"	790'-10"	221°45'		¾ in.	Reactor Pressure Level "B"
1MD-130	791'-3"	790'-10"	142°30'		¾ in.	Reactor Pressure Level "C"
1MD-131	791'-3"	790'-10"	313°		¾ in.	Reactor Pressure Level "D"
1MD-132	783'-3"	782'-10"	233°30'		¾ in.	Reactor Pressure Level "B"
1MD-133	783'-3"	782'-10"	142°30'		¾ in.	Reactor Pressure Level "C"
1MD-134	771'-3"	770'-10"	44°		¾ in.	Reactor Pressure Level "A"
1MD-135	771'-3"	770'-10"	189°30'		¾ in.	Reactor Pressure Level "B"
1MD-136	771'-3"	770'-10"	157°15'		¾ in.	Reactor Pressure Level "C"
1MD-137	771'-2-3/4"	770'-9-3/4"	318°30'		¾ in.	Reactor Pressure Level "D"
1MD-138						Not Used
1MD-139						Not Used
1MD-140						Not Used
1MD-141	748 ft-0 in.	748 ft-0 in.	20°		8 in.	Spare (Capped)
1MD-142	748 ft-0 in.	748 ft-0 in.	292°		8 in.	Spare (Capped)
1MD-143						Not Used
1MD-144						Not Used
1MD-145						Not Used
1MD-146						Not Used
1MD-147	748 ft-0 in.	748 ft-0 in.	30°		12 in.	Spare (Capped)
1MD-148	748 ft-0 in.	748 ft-0 in.	270°		12 in.	Inservice Inspection Cables
1MD-150	784 ft-11 in.	784 ft-6 in.	135°		¾ in.	Instrumentation
1MD-151	778 ft-6 in.	778 ft-1 in.	115°		¾ in.	Spare (Capped)
1MD-152	765 ft-6 in.	765 ft-1 in.	65°	+17 ft-3 in.	¾ in.	Instrumentation
1MD-153	782 ft-4 in.	781 ft-11 in.	222°		¾ in.	Spare (Capped)
1MD-154	775 ft-11 in.	775 ft-6 in.	197°30'		¾ in.	Containment Monitoring System
1MD-155	763 ft-4 in.	762 ft-11 in.	167°	+17 ft-3 in.	¾ in.	Instrumentation
1MD-156	781 ft-11 in.	781 ft-6 in.	320°30'		¾ in.	Spare (Capped)
1MD-157	774 ft-11 in.	774 ft-0 in.	314°		¾ in.	Spare (Capped)
1MD-158	763 ft-4 in.	762 ft-11 in.	288°	+17 ft-3 in.	¾ in.	Spare (Capped)
1MD-159	784 ft-11 in.	784 ft-6 in.	40°		¾ in.	Reactor Press. Vessel Level "A"
1MD-160	778 ft-6 in.	778 ft-1 in.	40°		¾ in.	Reactor Press. Vessel Level "A"
1MD-161	765 ft-6 in.	765 ft-1 in.	35°	+17 ft-3 in.	¾ in.	Main Steam "A"/"C" & R.C.I.C. Steam
1MD-162						Not Used
1MD-163	744 ft-0 in.	744 ft-5 in.	173°		¾ in.	Drywell Pressure
1MD-164	744 ft-0 in.	744 ft-5 in.	179°		¾ in.	Recirc. Pump "A" Flow/Leak Detection (1E31-N764)

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

DRYWELL MECHANICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1MD-165	750 ft-0 in.	750 ft-5 in.	188°		¾ in.	Instrumentation
1MD-166	744ft-0 in.	743ft-7 in.	269°		¾ in.	Spare (Capped)
1MD-167	768ft-0 in.	768ft-6 in.	280°	+17ft-3 in.	¾ in.	Instrumentation
1MD-168	765ft-0 in.	765ft-6 in.	28°	+17ft-3 in.	¾ in.	Instrumentation
1MD-169	765ft-0 in.	764ft-6 in.	302°	-17ft-3 in.	¾ in.	HPCS Leak Detection
1MD-170	744ft-0 in.	744ft-5 in.	45°		¾ in.	Instrumentation
1MD-171	744ft-0 in.	744ft-5 in.	75°		¾ in.	Reactor Water Cleanup Flow
1MD-172						Not used
1MD-173	744ft-0 in.	743ft-7 in.	155°		¾ in.	Instrumentation
1MD-174	771ft-0 in.	770ft-6 in.	272°	-17ft-3 in.	1-1/2 in.	Spare (Capped)
1MD-175	747ft-0 in.	747ft-5 in.	264°		¾ in.	Instrumentation
1MD-176	785ft-6 in.	785ft-1 in.	256°		¾ in.	Instrumentation
1MD-177	747ft-0 in.	747ft-5 in.	188°		¾ in.	Recirc. Pump “A” Flow
1MD-178	744ft-0 in.	744ft-5 in.	290°		¾ in.	Drywell Pressure
1MD-179	764ft-6 in.	764ft-0 in.	272°	-17ft-3 in.	¾ in.	Instrumentation
1MD-180	744ft-0 in.	744ft-5 in.	295°		¾ in.	Recirc. Pump “B” Flow
1MD-181	747ft-0 in.	747ft-5 in.	155°		¾ in.	Recirc. Pump “A” #1 & 2 Seal Cavity
1MD-182	765ft-0 in.	764ft-6 in.	60°	+17ft-3 in.	¾ in.	RI/RH Leak Detection System
1MD-183	764ft-0 in.	763ft-6 in.	152°	-17ft-3 in.	¾ in.	Instrumentation
1MD-184	764ft-0 in.	763ft-6 in.	158°	-17ft-3 in.	¾ in.	Spare (Capped)
1MD-185	765ft-0 in.	764ft-6 in.	236°	+17ft-3 in.	¾ in.	Main Steam Line “D”
1MD-186	765ft-ft-0 in.	764ft-6 in.	274°	+17ft-3 in.	¾ in.	Main Steam Flow
1MD-187	747ft-0 in.	746ft-7 in.	145°		¾ in.	Instrumentation
1MD-188	749ft-0 in.	749ft-6 in.	295°		¾ in.	Recirc. Pump “B” #1 & 2 Seal Cavity
1MD-189	744ft-0 in.	744ft-5 in.	276°		¾ in.	Spare (Capped)
1MD-190	771ft-0 in.	771ft-6 in.	173°	-17ft-3 in.	¾ in.	Spare (Capped)
1MD-191	771ft-0 in.	771ft-6 in.	212°	-17ft-3 in.	¾ in.	Spare (Capped)
1MD-192	761ft-6 in.	761ft-0 in.	91°		¾ in.	Instrumentation
1MD-193	771ft-ft-0 in.	771ft-6 in.	134°	-17ft-3 in.	¾ in.	Spare (Capped)
1MD-194	771ft-0 in.	771ft-6 in.	167°	-17ft-3 in.	¾ in.	Containment Monitoring System
1MD-195	768ft-0 in.	768ft-6 in.	173°	-17ft-3 in.	¾ in.	Spare (Capped)
1MD-196	771ft-0 in.	770ft-6 in.	180°	-17ft-3 in.	1 in.	Spare (Capped)
1MD-197	765ft-0 in.	764ft-6 in.	120°	+17ft-3 in.	¾ in.	Spare (Capped)
1MD-198	751ft-0 in.	750ft-7 in.	155°		¾ in.	Spare (Capped)
1MD-199	771ft-0 in.	770ft-6 in.	287°		1 in.	Spare (Capped)
1MD-200	765ft-ft-0 in.	764ft-6 in.	208°	+17ft-3 in.	¾ in.	Main Steam (Loop “A”) – RHR (Loop “C”)
1MD-201	750ft-0 in.	749ft-6 in.	262°		¾ in.	Instrumentation
1MD-202						Not used
1MD-203	747ft-0 in.	747ft-7 in.	340°		¾ in.	Spare (Capped)
1MD-204	747ft-5 in.	747ft-0 in.	355°		¾ in.	Instrumentation
1MD-205	768ft-0 in.	768ft-6 in.	270°	+17ft-3 in.	¾ in.	Drywell Pressure
1MD-206	761ft-6 in.	761ft-0 in.	57°	+17ft-3 in.	3/4 in.	Spare (Capped)

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

DRYWELL MECHANICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1MD-207	750ft-0 in.	749ft-6 in.	75°		3/4 in.	Spare (Capped)
1MD-208	777ft-0 in.	777ft-6 in.	165°		3/4 in.	Spare (Capped)
1MD-209	774ft-0 in.	774ft-6 in.	155°		3/4 in.	Spare (Capped)
1MD-210	777ft-0 in.	777ft-6 in.	155°		3/4 in.	Spare (Capped)
1MD-211	777ft-0 in.	777ft-6 in.	170°		3/4 in.	Spare (Capped)

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

DRYWELL ELECTRICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1ED-1	791ft- 10-23/32 in.	791ft- 10-23/32 in.	62.5°		6 in.	Neutron Monitoring 1K1N
1ED-2	791ft- 11-1/32 in	791ft- 11-1/32 in.	64°		6 in.	Neutron Monitoring 1K1N
1ED-3	793ft-3 in.	793ft-3 in.	62.5°		4 in.	Instrumentation 1K1E
1ED-4	793ft-3 in.	793ft-3 in.	63.5°		4 in.	Instrumentation 1K1E
1ED-5	775ft-0 in.	775ft-0 in.	36°		6 in.	Medium Voltage 6900 V Recirc. Pump 1P1B
1ED-6	775ft-0 in.	775ft-0 in.	37.5°		6 in.	Medium Voltage 6900 V Recirc. Pump 1P1B
1ED-7	774ft-0 in.	774ft-0 in.	44°		4 in.	Instrumentation 1K1B
1ED-8	774ft-0 in.	774ft-0 in.	45°		4 in.	Instrumentation 1K1B
1ED-9	794ft-9 in.	794ft-9 in.	62.5°		4 in.	Control 1C1E
1ED-10	794ft-9 in.	794ft-9 in.	63.5°		4 in.	Control 1C1E
1ED-11	796ft-3 in.	796ft-3 in.	62.5°		4 in.	L.V. Power 1P1E
1ED-12	796ft-3 in.	796ft-3 in.	63.5°		4 in.	L.V. Power 1P1E
1ED-13	796ft-3 in.	796ft-3 in.	64.5°		4 in.	L.V. Power 1P1E
1ED-14	774ft-0 in.	774ft-0 in.	52°		4 in.	L.V. Power 1P1B
1ED-15	774ft-0 in.	774ft-0 in.	53°		4 in.	L.V. Power 1P1B
1ED-16	774ft-0 in.	774ft-0 in.	54°		4 in.	L.V. Power 1P1B
1ED-17	774ft-0 in.	774ft-0 in.	48°		4 in.	Control 1C1B
1ED-18	774ft-0 in.	774ft-0 in.	49°		4 in.	Control 1C1B
1ED-19	774ft-9 in.	774ft-9 in.	48°		4 in.	Control 1C1B
1ED-20	774ft-9 in.	774ft-9 in.	49°		4 in.	Control 1C1B
1ED-21	773ft-9 in.	773ft-9 in.	135°	-11ft- 4-7/32 in.	4 in.	Control 1C3E
1ED-22	773ft-9 in.	773ft-9 in.	135°	-11ft- 11-19/32 in.	4 in.	Control 1C3E
1ED-23	773ft-9 in.	773 ft-9 in.	135°	-12ft- 6-15/16 in.	4 in.	Control 1C3E
1ED-24	792ft-6 in.	792ft-6 in.	157°	-13ft- 1-1/16 in.	6 in.	Neutron Monitoring 1K3N
1ED-25	792ft-6 in.	792ft-6 in.	157°	-12ft- 3-1/16 in.	6 in.	Neutron Monitoring 1K3N
1ED-27	773ft-9 in.	773ft-9 in.	135°	-13ft- 1-19/32 in.	4 in.	Control 1C3E
1ED-28	773ft-9 in.	773ft-9 in.	135°	-13ft- 9-1/8 in.	4 in.	Control 1C3E
1ED-29	773ft-9 in.	773ft-9 in.	203°		6 in.	Instrumentation 1K2E
1ED-30	773ft-9 in.	773ft-9 in.	205°		6 in.	Instrumentation 1K2E
1ED-31	776ft-3 in.	776ft-3 in.	203°		4 in.	Control 1C2E
1ED-32	776ft-3 in.	776ft-3 in.	204°		4 in.	Control 1C2E
1ED-33	776ft-3 in.	776ft-3 in.	205°		4 in.	Control 1C2E
1ED-34	774ft- 2-3/8 in.	774ft- 2-3/8 in.	242°		4 in.	L.V. Power 1P2E
1ED-35	774ft- 2-5/32 in.	774ft- 2-5/32 in.	243°		4 in.	L.V. Power 1P2E
1ED-36	774ft- 2-3/16 in.	774ft- 2-3/16 in.	244°		4 in.	L.V. Power 1P2E
1ED-37	773ft-9 in.	773ft-9 in.	300.5°		4 in.	Instrumentation 1K2B
1ED-38	773ft-9 in.	773ft-9 in.	301.5°		4 in.	Instrumentation 1K2B
1ED-39	773 ft-9 in.	773ft-9 in.	302.5°		4 in.	Instrumentation 1K2B
1ED-40	773ft-9 in.	773ft-9 in.	303.5°		4 in.	Instrumentation 1K2B
1ED-41	773ft-9 in.	773ft-9 in.	304.5°		4 in.	Instrumentation 1K2B
1ED-43	774ft-0 in.	774ft-0 in.	310.5°		4 in.	Control 1C2B

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

DRYWELL ELECTRICAL PENETRATIONS							
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description	
1ED-44	774ft-0 in.	774ft-0 in.	311.5°		4 in.	Control	1C2B
1ED-45	774ft-0 in.	774ft-0 in.	312.5°		4 in.	Control	1C2B
1ED-46	774ft-0 in.	774ft-0 in.	306.5°		4 in.	L.V. Power	1P2B
1ED-47	774ft-0 in.	774ft-0 in.	307.5°		4 in.	L.V. Power	1P2B
1ED-48	774ft-0 in.	774ft-0 in.	308.5°		4 in.	L.V. Power	1P2B
1ED-49	773ft-9 in.	773ft-9 in.	296.5°		6 in.	Spare	
1ED-50	773ft-9 in.	773ft-9 in.	298°		6 in.	Control	1C2E
1ED-51	774ft-0 in.	774ft-0 in.	315.5°		6 in.	Neutron Monitoring	1K4N
1ED-52	774ft-0 in.	774ft-0 in.	317°		6 in.	Neutron Monitoring	1K4N
1ED-53	774ft-0 in.	774ft-0 in.	318.5°		4 in.	Instrumentation	1K4E
1ED-54	774ft-0 in.	774ft-0 in.	320°		4 in.	Instrumentation	1K4E
1ED-55	774ft-0 in.	774ft-0 in.	322°		4 in.	Control	1C4E
1ED-56	774ft-0 in.	774ft-0 in.	323°		4 in.	Control	1C4E
1ED-57	774ft-0 in.	774ft-0 in.	325°		4 in.	Control	1C4E
1ED-58	773ft-9 in.	773ft-9 in.	135°	-16ft-6-3/32 in.	4 in.	Instrumentation	1K3E
1ED-59	773ft-9 in.	773ft-9 in.	135°	-15ft-8-3/8 in.	4 in.	Instrumentation	1K3E
1ED-60	774ft-9 in.	774ft-9 in.	203°		4 in.	Control	1C2E
1ED-61	774ft-9 in.	774ft-9 in.	205°		4 in.	Neutron Monitoring	1K2N
1ED-64	792ft-0 in.	792ft-0 in.	103.5°		6 in.	Control Rod Drive Position Indication	1K1E
1ED-65	792ft-0 in.	792ft-0 in.	105°		6 in.	Control Rod Drive Position Indication	1K1E
1ED-66	792ft-0 in.	792ft-0 in.	106.5°		6 in.	Control Rod Drive Position Indication	1K1E
1ED-67	792ft-0 in.	792ft-0 in.	108°		6 in.	Control Rod Drive Position Indication	1K1E
1ED-68	794ft-3 in.	794ft-3 in.	303°		6 in.	Control Rod Drive Position Indication	1K2E
1ED-69	794ft-3 in.	794ft-3 in.	304.5°		6 in.	Control Rod Drive Position Indication	1K2E
1ED-70	794ft-3 in.	794ft-3 in.	306°		6 in.	Control Rod Drive Position Indication	1K2E
1ED-71	794ft-3 in.	794ft-3 in.	307.5°		6 in.	Control Rod Drive Position Indication	1K2E
1LD-1	751ft-0 in.	751ft-0 in.	47°		3 in.	Ltg./Comm.	
1LD-2	751ft-0 in.	751ft-0 in.	48°		3 in.	Ltg./Comm.	
1LD-3	751ft-0in.	751ft-0 in.	49°		3 in.	Ltg./Comm.	
1LD-4	750ft-0 in.	750ft-0 in.	273°		3 in.	Ltg./Comm.	
1LD-5	750ft-0 in.	750ft-0 in.	274°		3 in.	Ltg./Comm.	
1LD-6	750ft-0 in.	750ft-0 in.	275°		3 in.	Ltg./Comm.	
1LD-7	775ft-0 in.	775ft-0 in.	57°		3 in.	Ltg./Comm.	
1LD-8	775ft-0 in.	775ft-0 in.	58°		3 in.	Ltg./Comm.	
1LD-9	743ft-0 in.	743ft-0 in.	5°		3 in.	Ltg./Comm.	
1LD-10	774ft-0 in.	774ft-0 in.	57°		3 in.	Ltg./Comm.	
1LD-11	774ft-0 in.	774ft-0 in.	58°		3 in.	Ltg./Comm.	
1LD-12	739ft-3 in.	739ft-3 in.	142°		3 in.	Ltg./Comm.	
1LD-13	742ft-6 in.	742ft-6 in.	178°		3 in.	Ltg./Comm.	
1LD-14	784ft-0 in.	784ft-0 in.	58°		3 in.	Ltg./Comm.	
1LD-15	774ft-0 in.	774ft-0 in.	279°		3 in.	Ltg./Comm.	

TABLE 3.8-5
CONTAINMENT AND DRYWELL PENETRATIONS (ContInued)

DRYWELL ELECTRICAL PENETRATIONS						
No.	Centerline Elevation Inside	Centerline Elevation Outside	Azimuth	Distance Parallel to Azimuth	Nom. Pipe Size	Description
1LD-16	774ft-0 in.	774ft-0 in.	280°		3 in.	Ltg./Comm.
1LD-17	774ft-0 in.	774ft-0 in.	281°		3 in.	Ltg./Comm.
1LD-18	774ft-0 in.	774ft-0 in.	282°		3 in.	Ltg./Comm.

TABLE 3.8-6
PREDICTED DEFORMATION OF THE
CONTAINMENT DURING THE PRESSURE TEST*

ITEM	LOCATION	METER NO.**	DEFLECTION (IN)
Radial deflection of cylinder wall	10 feet above base	RC1-RC6	0.16
	48 feet above base	RC7-RC12	0.43
	Midheight of cylinder	RC13-RC18	0.43
	123 feet above base	DC19-DC21	0.86***
	Dome springline	DC22-DC24	0.72***
	Equipment hatch	RHC25-RHC36	0.43
Vertical deflection of dome with reference to cylinder base	Dome springline	VC1-VC6	0.25
	1st intermediate point (at El. 909 ft)	VC7	0.42
	2nd intermediate point (at El. 920 ft)	VC8	0.45
	Dome apex	VC9	0.45

* Test pressure is 17.25 psig.
** The locations of the deflection meters are shown on Drawing S27-1401.
*** The value given is the diametrical deflection of the dome springline.

TABLE 3.8-7.1
PREDICTED DEFORMATION OF
THE DRYWELL FOR THE
PRESSURE TEST*

ITEM	LOCATION	METER NO.**	DEFLECTION (in.)
Radial deflection of drywall wall	10 ft. above base	RD1 - RD3	0.105
	48 ft. above base	RD4 - RD6	0.254
	Top of drywell	RD7 - RD9	0.098
	Equipment hatch region	RHD10 - RHD21	0.238
Vertical deflection	Base to top of drywell	VD1 - VD3	0.250
	Top of shield wall to top of drywell	VD4 - VD6	0.800

* The test pressure is 30.0 psig.

**The locations of the deflection meters are shown on Drawing S27-1401.

TABLE 3.8-7.2
PREDICTED STRAINS OF THE DRYWELL FOR THE PRESSURE TEST*

ITEM	LOCATION	GAUGE NO.**	STRAIN	
			INSIDE FACE	OUTSIDE FACE
Hoop strain of drywell wall	10 feet above base	SD1 - SD2	0.00027	0.00027
	48 feet above base	SD3 - SD4	0.00059	0.00062
Vertical strain of drywell wall	10 feet above base	SD1 - SD2	0.00007	0.00001
	48 feet above base	SD3 - SD4	0.00002	0.00004

* The test pressure is 30.0 psig.

**The locations of the strain gauges are shown in Drawing S27-1401.

TABLE 3.8-8
MATERIAL PROPERTIES FOR CONTAINMENT ULTIMATE CAPACITY STUDY

Material	Specified Strength	Average Tested Strength	Standard Deviation
Concrete	4,000 psi	6,086 psi	
#11 rein. steel	60 ksi	68.6 ksi	3.3 ksi
#14 rein. steel	60 ksi	73.3 ksi	2.9 ksi
#18 rein. steel	60 ksi	71.1 ksi	2.6 ksi
Carbon steel liner (1/4 in.)	32 ksi	48.4 ksi	3.3 ksi
Stainless steel liner (1/4 in. and 1/2 in.)	30 ksi	43.0 ksi	3.7 ksi

TABLE 3.8-9
CODES USED FOR DESIGN AND CONSTRUCTION OF
STRUCTURAL ITEMS INSIDE CONTAINMENT[1,3]

ITEM	DESIGN	MATERIAL		FABRICATION		ERECTION		ANI	STAMPING
		SPECIFICATION	CONTROL[2]	WELDING	EXAMINATION	WELDING	EXAMINATION		
1. CONTAINMENT									
a. Liner backed by concrete	ASME, (proposed) Section III, Division 2 (1973)	ASME, Section II (1971,Sum.'73)	CC-2500, ASME, Sec. III, Div. 2 (Proposed-1973)	CC-4500, ASME, Sec. III, Div. 2 (Proposed-1973) and ASME, Section IX (1971, Summer '73)	NRC Regulatory Guide 1.19 Rev. 1 and NE-5000, ASME, Sec. III, Div. 1 (1971, Sum. '73)	CC-4500,ASME, Section III, Div. 2 (Proposed - 1973) and ASME, Section IX (1971, Summer '73)	NRC Regulatory Guide 1.19, Rev. 1 and NE-5000, ASME, Sec. III, Div. 1 (1971, Sum. '73)	No	No
b. Pipe Penetration Sleeves	ASME, Section III Division 1 (1974 with Summer '74 Addenda), Sub-section NE	ASME, Section II (1971, Sum. '73)	NE-2000 ASME, Sec. III, Div. 1 (1971, Sum. '73)	ASME, Section IX (1971, Summer '73)	NE-2000, ASME, Sec. III, Div. 1 (1971, Sum. '73)	NE-4000, ASME, Sec. III, Div. 1 (1971, Summer '73)	NE-5000, ASME, Sec. III, Div. 1 (1971, Summer '73)	Yes	No
c. Personnel Locks	ASME, Section III Division 1 (1971 with Summer '73 Addenda), Sub-section NE	ASME, Section II (1971, Summer '73)	NA-4000 & NE-2000, ASME, Sec. III, Div. 1 (1971, Sum. '73)	NE-4000, ASME, Sec. III, Div. 1 (1971, Sum. '73) and ASME, Section IX (1971, Sum. '73)	NE-5000, ASME, Sec. III, Div. 1 (1971, Sum. '73)	NE-4000, ASME, Sec. III, Div. 1 (1971, Sum. '73) and ASME, Section IX (1971, Sum. '73)	NE-5000, ASME, Sec. III, Div. 1 (1971, Sum. '73)	Yes	Yes
d. Equipment Hatch	ASME, Section III Division 1 (1971 with Summer '73 Addenda), Sub-section NE	ASME, Section II (1971, Summer '73)	NA-4000 & NE-2000, ASME, Sec. III, Div. 1 (1971, Sum. '73)	NE-4000, ASME, Sec. III, Div. 1 (1971, Sum. '73) and ASME, Section IX (1971, Sum. '73)	NE-5000, ASME, Sec. III, Div. 1 (1971, Sum. '73)	NE-4000, ASME, Sec. III, Div. 1 (1971, Sum. '73) and ASME, Section IX (1971, Sum. '73)	NE-5000, ASME Sec. III, Div. 1 (1971, Sum. '73)	Yes	No
2. DRYWELL									
a. Suppression Pool Liner (Backed by Concrete)	ASME, (Proposed) Section III, Division 2 (1973)	ASME, Section II (1974, Winter '75)	Equivalent to NB-2000 & NA-4000, ASME, Sec. III, Div. 1 (1974 Edition, Summer '74)	Equivalent to NB-4000, ASME, Sec. III, Div. 1 (1974, Summer '74) and ASME Section IX (1974 Edition, Winter 1975) or AWS D1.1, Rev. 1-76	NB-5000, ASME, Sec. III, Div. 1 (1974, Win. '75) and ASME, Section V (1974 Edition, Winter '75)	NB-4000, ASME, Sec. III, Div. 1 (1974, Summer '74) and ASME, Section IX (1974, Summer '74) or AWS D1.1, Rev. 1-'76	NB-5000, ASME, Sec. III, Div. 1 (1974, Summer '75) and ASME, Section V (1974, Summer '75)	No	No

TABLE 3.8-9
CODES USED FOR DESIGN AND CONSTRUCTION OF
STRUCTURAL ITEMS INSIDE CONTAINMENT[1,3] (Continued)

	DESIGN	MATERIAL		FABRICATION		ERECTION		ANI	STAMPING
ITEM		SPECIFICATION	CONTROL[2]	WELDING	EXAMINATION	WELDING	EXAMINATION		
b. Form Plate	AISC 1969 or 1978	ASTM	Certified Material Test Reports Only	ASME, Sec. IX (1974, Win. '75) OR AWS D1.1, Rev 1-'76	Not Applicable	ASME, Section IX (1974 Edition, Summer '74) OR AWS D1.1-Rev. 1, 1976	NB-5000, ASME, Sec. III, Div. 1 (1974 Edition, Summer '75) and ASME, Section V (1974, Summer '75)	No	No
c. Pipe Penetration Sleeves	ASME, Section III, Div. 1 (1974, Summer 1974), Subsection NF	ASME, Section II (1974, Sum. '74)	NF-2000& NA-4000, ASME, Section III, Div. 1 (1974, Sum. '74)	ASME, Sec. III (1974, Sum. '74) AND ASME, Section IX (1974, Sum. '74)	NF-2000, ASME, Section III, Division 1 (1974, Sum. '74)	ASME, Section IX (1974, Sum. '74) OR AWS D1.1, Rev. 1-'76	NB-5000, ASME, Sec. III, Div. 1 (1974, Sum. '75) and ASME, Section V (1974, Sum. '75)	No	No
d. Personnel Lock & Hatch, Drywell Head	ASME, Section III, Division 1, Sub-section NE, (1974 Edition, Summer '76 Addenda)	ASME, Section II (1974, Summer '76)	NE-2000 & NA-4000, ASME, Sec. III, Div. 1 (1974, Sum. '76)	NE-4000, ASME Sec. III, Div. 1 (1974, Sum. '76) and ASME, Section IX (1974, Sum. '76)	NE-5000, ASME Sec. III, Div. 1 (1974, Sum. '76) and ASME, Sec. V (1974, Sum. '76)	NE-4000, ASME, Sec. III, Div. 1 (1974, Sum. '76 and ASME, Sec. IX (1974, Sum. '76) or AWS D1.1, Rev. 1-'76	NB-5000, ASME, Sec. III, Div. 1 (1974, Sum. '75) and ASME, Section V (1974, Sum. '76)	Yes	No
e. Refueling Bellows	ASME, Section III, Division 1, Sub-section NE, (1977 Edition, Winter '77 Addenda)	ASME, Sec. II, (1977, Winter '77)	NE-2000 & NA-4000, ASME, Sec. III, Div. 1 (1977, Winter '77)	NE-4000, ASME, Sec. III, Div. 1 (1977, Win. '77) and ASME, Section IX (1977, Win. '77)	NE-5000, ASME, Sec. III, Div. 1 (1977, Win. '77) and ASME, Section V (1977, Win. '77)	NE-4000, ASME, Sec. III, Div. 1 (1974, Sum. '74) and ASME, Section IX (1974, Sum. '74)	NE-5000, ASME, Sec. III, Div. 1 (1974, Sum. '74) and ASME, Section V (1974, Sum. '74)	No	No
3. REACTOR PEDESTAL	ASME, Section III, Division 1, Sub-section NF (1974 Edition, Winter '75 Addenda)	ASME, Sec. II (1974, Winter '75)	NA-4000, NF-2000, ASME, Section III, Div. 1 (1974, Win. '75)	NF-4000, ASME, Sec. III, Div. 1 (1974, Win. '75) and ASME, Section IX (1974, Winter '75)	NF-5000, ASME, Sec. III, Div. 1 (1974, Win. '75) and ASME, Section V (1974, Win. '75)	NF-4000, ASME, Sec. III, Div. 1 (1974, Win. '75) and ASME Sec. IX (1974, Win. '75)	NF-5000, ASME, Sec. III, Div. 1 (1974, Win. '75) and ASME, Section V (1974, Win. '75)	No	No

4.	WEIR WALL LINER	ASME, (Proposed) Section III, Division 2 (1973)	ASME, Sec. II, (1974, Win. '75)	Equivalent to NB-2000 & NA-4000, ASME, Sec. III, Div. 1 (1974, Sum. '74)	ASME, Section IX (1974, Win. '75) OR AWS D1.1, Rev. 1-1976	Equivalent to CC-5500, ASME, Sec. III, Div. 2 (Proposed, 1973) and NB-5000, ASME, Sec. III, Div.1 (1974, Sum. '74)	ASME, Section IX (1974, Sum. '74) OR AWS D1.1, Rev. 1-1976	Equivalent to CC-5500, ASME, Sec. III, Div. 2 (Proposed, 1973) and NB-5000, ASME, Sec. III, Div. 1 (1974, Sum. '75) and ASME Sec. V (1974, Sum. ‘75)	No	No
5.	REACTOR SHIELD WALL	AISC 1969 or 1978	ASTM	Equivalent to NB-2500 & NA-4000, ASME, Sec. III, Div. 1 (1974, Win. '75)	ASME, Section IX (1974, Winter '75) OR AWS D1.1, Rev. 1-1976	NB-5000, ASME Sec. III, Div.1 (1974, Win. '75) and ASME, Section V (1974, Win. '75)	ASME, Section IX (1974, Sum. '74) OR AWS D1.1, Rev. 1-1976	NB-5000, ASME, Sec. III, Div. 1 (1974, Sum. '75) and ASME, Section V (1974, Sum. '75)	No	No
6.	CONTAINMENT POOL									
	a. Stainless Steel Liner	ASME, Section III Division 2 (1977)	ASME, Sec. II (1977 Edition)	Equivalent to NB-2000 & NA-4000, ASME, Sec. III, Div. 1 (1977)	CC-4000, ASME, Sec. III, Div. 2 (1977) and ASME, Section IX (1977)	Equivalent to CC-5500, ASME, (1977) and ASME, Section V (1977)	CC-4000, ASME, Sec. III, Div. 2 and ASME, Section IX (1977)	Equivalent to CC-5500, ASME, (1977) and ASME, Section V (1977)	No	No
	b. Pool Gates	ASME, Section III Division 1 (1977) Subsection ND	ASME, Sec. II (1977 Edition)	ND-2000 & NA 4000, ASME, Section III, Division 1 (1977)	ND-4000, ASME, Sec. III, Div. 1 (1977) and ASME, Section IX (1977)	Equivalent to CC-5500, ASME, Sec. III, Div. 2 (1977) and ASME, Section V (1977)	ND-4000, ASME, Sec. III, Div. 1 (1977) and ASME Sec. IX (1977)	Equivalent to CC-5500, ASME, Sec. III, Div. 2 (1977) and ASME, Section V (1977)	No	No
7.	PIPE WHIP RESTRAINTS	AISC (1969 or 1978)	ASTM	NF-2000 & NA-4000, ASME, Sec. III, Div. 1 (1977, Win. '78)	NF-4000, ASME, Sec. III, Div. 1 (1974, Win. '75) and ASME, Section IX (1974, Win. '75)	Equivalent to NF-5000, ASME, Sec. III, Div. 1 (1974, Win. '75) and ASME, Sec. V (1974, Win. '75)	NF-4000, ASME, Sec. III, Div.1 (1974, Winter '75) and ASME, Section IX (1974, Winter '75)	Equivalent to NF-5000, ASME, Sec. III, Div. 1 (1974, Winter '75) and ASME, Section V (1974, Win. '75)	No	No
8.	CONCRETE AND REBAR									
	a. Concrete (Includes containment wall, drywell wall, weir wall, contain-	ASME, (Proposed) Section III, Division 2 (1973)	ACI Codes and Standards (Refer to Table 3.8-4)	ANSI N45.2.5, Draft 3, Rev. 1 Jan. 1974	Not Applicable	Not Applicable	Not Applicable	Not Applicable	No	No

TABLE 3.8-9
CODES USED FOR DESIGN AND CONSTRUCTION OF
STRUCTURAL ITEMS INSIDE CONTAINMENT[1,3] (Continued)

	ment pool walls & refueling floor slab)									
b.	Rebar (Includes same items as above)	ASME, (Proposed) Section III, Division 2 (1973)	ASTM A615 (Refer to Table 3.8-4)	Certified Material Test Reports Only	CRSI (Refer to Table 3.8-4)	CRSI (Refer to Table 3.8-4)	NRC Regulatory Guide 1.10, Rev. 1 and CC 4300, ASME Sec. III, Div. 2 (1975)	NRC Regulatory Guide 1.10, Rev. 1 and CC 4300, ASME ec. III, Div. 2 (1975)	No	No

NOTES:

1. References to the ASME Code paragraphs made throughout this table are for technical requirements only; administrative requirements, such as preparation of Design Specification, Design Report, Code Data Report, ANI involvement or stamping, are not included unless noted in last two columns.

2. Material Control includes requirements for Certified Material Test Reports, impacts, identification & traceability, unless otherwise indicated.

3. Table contents may include non-code reference information.

ATTACHMENT A3.8
CONTAINMENT DESIGN LOADS

ATTACHMENT A3.8 - CONTAINMENT DESIGN LOADS

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CONTAINMENT DESIGN LOADS

A3.8.1 INTRODUCTION

The methodologies used to determine the design-basis loads for structural components for the Clinton Power Station (CPS) are presented herein. The methodology for each loading phenomenon is discussed in detail and the critical load combinations and acceptance criteria are identified. Finally, the system analysis method is discussed.

Section A3.8.2 discusses the methodology for determining safety-relief valve (SRV) actuation loads as they apply to the suppression pool boundaries and submerged structures.

Section A3.8.3 discusses the loads resulting from a loss-of-coolant accident (LOCA). Each LOCA-related phenomenon is identified and the loading methodology is presented.

Section A3.8.4 identifies the remaining loads that act on the structure i.e., normal, seismic, thermal, etc. Section A3.8.5 identifies the load combinations and acceptance criteria, and Section A3.8.6 presents the analysis methods used to determine structural responses and the acceptance of the structural design.

A3.8.2 DEVELOPMENT OF SRV LOADS

The structural design basis for the Clinton Power Station was based on the assumption that rams head discharge devices would be installed on the SRV discharge lines. This assumption was consistent with the technical understanding of the phenomena and licensing at the time the construction permit was issued (see Paragraph 5, pp.6-9 of Reference 1.) Since that time, General Electric (GE) has determined that the quencher discharge device is a desirable alternative to the rams head device in that it substantially reduces the suppression pool boundary loads resulting from air-clearing phenomena during SRV discharge and minimizes thermal effects in the suppression pool. Further, GE has specified the quencher device for the Standard BWR/6-238 design and recommended the device for BWR/6-Mark III application (see Attachment A of Reference 2). The quencher device has been incorporated into the CPS design. Therefore, this section presents the methodologies used for determining the structural design-basis loads for both the devices. The former are presented to show compliance with the licensing commitments described in Reference 1, and the latter are presented to demonstrate the compatibility of the design with the quencher discharge device loads.

The design-basis SRV loads on BOP piping and equipment are presented in Attachment A3.9.

A3.8.2.1 Description of the Phenomena

Prior to the actuation of a pressure relief valve, the piping between the SRV discharge and the suppression pool water surface is full of air at drywell pressure and temperature. The discharge piping terminates at the quencher in the suppression pool, with the water level inside the pipe at the same level as the water at the pool surface.

When a relief valve lifts, the effluent reactor steam causes a rapid pressure buildup in the discharge pipe. This results in a rapid compression of the column of air in the discharge pipe and acceleration of the water in the submerged portion of the piping and the expulsion of the water through the line. The pressure in the pipe builds to a peak as the last of the water is expelled. The compressed cushion of air between the water slug and the steam exits through the quencher, forming a number of clouds of small bubbles which begin to expand to the lower pool pressure. These bubbles continue to expand, displacing the pool water and propagating a pressure disturbance throughout the suppression pool. When the gas pressure reaches equilibrium with the local hydrostatic pressure, the transient would cease were it not for the inertia of the accelerated water mass. The inertia of the water drives the gas system past the point of equilibrium, and a negative pressure (with respect to local hydrostatic pressure) results within the bubble. The negative pressure in the bubble decelerates the water mass and reverses its motion in an attempt to reach equilibrium. Again, the inertia of the water drives the system past the equilibrium pressure, and the process repeats in a cyclic manner. The dynamics of the air-water system are manifest in pressure oscillations (similar to that of a springmass system) arising from the expansion and contraction of the bubble coupled with the inertial effects of the moving water mass. The oscillations are repeated with an identifiable frequency until the bubble reaches the pool surface.

The magnitude of the pressure disturbance in the suppression pool decreases with increasing distance from the point of discharge, resulting in a damped oscillatory load at every point on the structures and pool boundaries below the pool surface.

A3.8.2.2 Analytical Models for Rams Head Loads

This section discusses the analytical models utilized to determine the bubble dynamics and wall loading data for discharge of an SRV through a rams head discharge device. The vent-clearing portion of the analysis is described in Reference 3.

It is the goal of this section to provide the basic background necessary to understand the analytical procedures and assumptions utilized in the design-basis determination.

These conclusions support the analytical models discussed in the following sections.

A3.8.2.2.1 Bubble Dynamics Model

The analytical model for the bubble dynamics is described in Subsections A3.8.3.1 and A3.8.3.4 and in Section 4.0 of Reference 3. The differential equations and initial conditions for bubble pressure, radius, and depth are summarized in Table A3.8-1. The analytical model includes the following features and assumptions:

a. Bubble formation efficiency, η, , accounts for energy lost by heat transfer to the pool; consequently, this lost energy is not present in the bubble, as explained in Reference 3. A value of 0.1 was used.

b. The environmental pressure ($P\infty$) includes the hydrostatic pressure in the pool which varies as the bubble rises. The reference pressure (P_{ref}) was assumed to be 14.7 psi above the pool surface. The oscillatory and vertical motions of the bubble are dependent upon the time-dependent description of the bubble depth, Z. Since this analysis includes the variation of bubble pressure with elevation, it is not advisable to assume that these effects are negligible as in Reference 3.

c. The vertical bubble motion description includes the effects of the bubble's virtual mass, drag, buoyancy, and preferential upward expansion. A drag coefficient (C_D) of 2.5 is used. The bubble radial displacement term ($R\Delta t$) is included only when the bubble is expanding ($R > 0$), representing preferential upward expansion of the bubble. The bubble is assumed to remain at the same initial depth until its formation is completed. Since both drag force and bubble mass are included in this analysis, it is not necessary to assume that these effects are negligible as in Reference 3.

d. The initial bubble pressure (P) is calculated as a stagnation pressure based on vent line discharge pressure, density, and velocity. Reference 3 uses the static pressure rather than the stagnation pressure as the initial bubble pressure. Since the stagnation pressure is larger than the static pressure, a higher (more conservative) initial bubble pressure is used. The discharge velocity is assumed to be sonic. The rams head geometry is taken into account in calculating discharge pressure and density by dividing by 2, assuming that two bubbles are formed. This results from the assumption that sonic velocity is established immediately upon clearing at both the end of the vent line and the rams head outlets.

e. The time required to discharge air (t_d) is based on the initial mass of air in the vent line and mass flow rate (m) based on discharge density and velocity. At this time (t_d), the first term in the pressure differential equation (Equation A3.8-8 in Table A3.8-1) is dropped and the bubble is allowed to rise.

f. The bubble dynamics calculation is terminated when the bubble surface reaches the pool surface.

A3.8.2.2.2 <u>Method of Images Model</u>

The method of images is a classical technique applied in many scientific and engineering disciplines. Hydrodynamic images have been utilized to describe the flow of an ideal fluid for over a century (see References 4 and 5). Usually, however, the fluid has been assumed to be infinite or semi-infinite in extent (see References 6 and 7). In order to describe the pressure field behavior in a finite fluid (such as a suppression pool), a particular interpretation of the method of hydrodynamic images was developed (Reference 3). Good agreement between the application of the method of images and a series of experiments has been reported (References 8 and 9). An experiment was designed and performed in order that the application of the method of hydrodynamic images to the fluid in a suppression pool could be verified. Verification of the principle of superposition with respect to the pressure field resulting from discharging of multiple SRV lines was also substantiated. As many of the normal operating conditions as possible were simulated in this scaled experiment. Based on the results of these experimental investigations, the method of hydrodynamic images is judged to be well suited for the suppression pool analyses.

Reference 3 presents the analytical models and equations that describe the bubble dynamics and the pool response. The development utilizes basic hydrodynamic potential flow theory (References 10 through 13), which assumes an ideal fluid. The bubble dynamics description is embodied in the Rayleigh equation, and the pool response description is derived from the generalized Bernoulli equation:

$$\frac{P}{\rho L} + \frac{1}{2}V^2 + \Omega - \frac{\partial\phi}{\partial t} = f(t) \qquad \text{(A3.8-1)}$$

where:

P	=	fluid pressure;
ρL	=	fluid density;
V	=	fluid velocity;
Ω	=	force potential;
$\varnothing$	=	velocity potential; and
f(t)	=	arbitrary function of time.

The velocity potential, , satisfies the LaPlace equation:

$$\nabla^2 \varnothing = 0 \tag{A3.8-2}$$

and it must also satisfy the initial and boundary conditions. The result is substituted into the generalized Bernoulli equation (A3.8-1), and an expression for the pressure distribution in the pool as a function of space and time is obtained. The boundary conditions which are imposed on the potential function are as follows:

a. There is no flow across the rigid boundaries.

b. The pressure at the free surface is constant and unaffected by the bubble.

c. At the bubble surface, the pressure is given as a function of time by the bubble dynamics equation.

The first two boundary conditions are satisfied through the use of the method of images. If image sources and sinks are placed as depicted in Figure A3.8-1, the effects of the rigid boundaries and the free surface can be described. The rigid boundaries are simulated by flow sources, and the free surface is simulated by sinks. The sources and sinks all have the same uniform strength.

The potential function must be harmonic to satisfy the LaPlace equation. For a simple point source, the following form is appropriate (Reference 12).

$$\varnothing = -\frac{M}{r} \tag{A3.8-3}$$

where:

M = source or sink strength, and

r = radial distance from the point source.

Since the sum of harmonic functions is also harmonic, the effect of all the simple point sources and sinks in the image array can be included through a simple summation. Thus, the local pressure (potential) in the suppression pool can be described by:

$$P - P_\infty = R_b \left[(P_b - P_\infty) + \rho L \frac{\dot{R}_b^2}{2\ gc} \right] \frac{\sum_{i=0}^{n_i} \frac{\pm 1}{r_{wi}}}{(1+R_b)\sum_{i=1}^{n_i} \frac{\pm 1}{r_{bi}}} \qquad \text{(A3.8-4)}$$

where:

ρL	=	liquid density;
P	=	local pressure;
$P\infty$	=	environmental pressure;
P_b	=	bubble pressure;
R_b	=	bubble radius;
$\dot{R}_b$	=	time rate of change of bubble radius;
r_{bi}	=	vector distance from the point of interest to the center of i^{th} source or sink;
r_{wi}	=	distance from the point on the boundary to the i^{th} image; i=0 for the bubble;
n	=	total number of real and imaginary sources and sinks; and
± 1	=	(+) depicts a source; (-) depicts a sink.

It should be noted that equation A3.8-4 is more general than the equivalent expression (Equation 35) given in Reference 3. Equation A3.8-4 applies during the entire bubble oscillation and not just at the extremes where R = 0. Hence, Equation A3.8-4 represents the total bubble pressure (i.e., the static and dynamic contributions are both included).

The third boundary condition is that the pressure at the bubble surface is equal to the internal pressure of the bubble.

This procedure accounts for the bubble dynamics, bubble rise, effects of the rigid and free boundaries, and pressure at the interface between the bubble and the suppression pool fluid.

The method was applied to the Mark III containment cylindrical geometry in all three dimensions, hence a general solution is embodied. The effects of the convex curvature of the drywell wall and concave curvature of the containment wall (relative to suppression pool fluid) are included in this general solution. Thus, the parametrics include the effects of pool geometry, bubble characteristics, and bubble location.

A3.8.2.2.3 Vent Clearing Model

The analytical model for the vent clearing transient model is fully described in Reference 3. This model has been extended to include the effect of friction on the water slug in the discharge line and implemented in the design calculations.

A3.8.2.2.4 Assumptions Used in the Implementation of the Analytical Model

In order to implement the analytical models described above, certain assumptions were required. These assumptions relate to assuring that conservative results are produced for use in design calculations. These assumptions are:

a. The actual plant geometry and piping configurations are used in the analysis.

b. The spring setpoints, Section III of the ASME Code, Article NB-7000, and additional conservative factors, to accommodate manufacturing tolerances, were used to determine the maximum individual SRV flow rates for use in the analysis.

c. The method of images was employed to account for the finite pool boundaries and size.

d. The loadings are determined by combining the effects of each bubble pair (two bubbles per rams head) for a given SRV discharge line using the SRSS method. Then the effects of each bubble pair were linearly summed.

e. The bubbles were located on the axis of the discharge device 4 feet from the exit plane of the device.

A3.8.2.2.5 Method of Implementation

In order that the loads are properly and accurately defined at the boundaries of the suppression pool as concisely as possible, the analytical models are implemented as follows:

a. The suppression pool vertical and horizontal boundaries are divided into three zones of equal length and the loading function is evaluated as a pressure at the midpoint of each zone for each boundary. (See Figure A3.8-4).

b. The pressure is determined as described above at azimuthal increments of 2° around the pool boundary.

c. The loading functions, i.e., the pressure distribution on each zone, are then represented as a Fourier Series. Thus, from Reference 14:

$$P(\theta) = \frac{Ao}{2} + \sum_{n=1}^{\infty} (A_n \cos n\theta + B_n \sin n\theta)$$

where:

$$A_n = \frac{1}{\pi} \int_{-\pi}^{\pi} P(\theta) \cos n\theta \, d\theta \qquad n = 0, 1, 2, \ldots$$

and

$$B_n = \frac{1}{\pi} \int_{-\pi}^{\pi} P(\theta) \sin n\theta \, d\theta \qquad n = 0, 1, 2, \ldots$$

Here,

$P(\theta)$ = the calculated pressure distribution on each zone,

A_n, B_n = the Fourier cosine and sine coefficients as defined above, and

θ = azimuth with the origin at the centerline of the RPV.

Each coefficient is then normalized by the maximum absolute value of each function over all nine zones. Therefore:

$$a_n = \frac{1}{\pi A^*_n} \int_{-\pi}^{\pi} P(\theta) \cos n\theta \, d\theta \qquad n = 0, 1, 2, \ldots$$

and
$$b_n = \frac{1}{\pi B^*_n} \int_{-\pi}^{\pi} P(\theta) \sin n\theta \, d\theta \qquad n = 0, 1, 2, 3 \ldots$$

and
$$p(\theta) = \frac{a_o}{2} + \sum_{n=1}^{\infty} (a_n \cos n\theta + b_n \sin n\theta).$$

where:

$p(\theta)$ is the normalized Fourier representation of the pressure function acting on each zone at any instant,

A_n* and B_n* are the maximum absolute values of the Fourier coefficients for all zones for all time, and

a_n and b_n are as defined above.

This process is repeated at each time step (typically 0.002 second) throughout the transient.

A3.8.2.2.6 Selection of Discharge Cases

The discharge cases selected for design purposes are those which give the maximum vertical and horizontal response for the containment and also the maximum overturning moment. These criteria are met by two discharge cases, the symmetric discharge case and the asymmetric case. These are discussed in the following.

A3.8.2.2.6.1 Symmetric Discharge Case

The symmetric discharge case is defined as the simultaneous actuation of all 16 SRV's.

A unique conservative bubble pressure time history is produced for each discharge line considering the valve setpoint, discharge line air mass, clearing time, discharge pressure, etc. These unique bubble dynamics are then considered for valves actuating simultaneously in the methodology described above. Figure A3.8-2 shows the distribution of the discharge devices in the suppression pool. Figure A3.8-4 shows a cross section of the pool which indicates the zones over which the boundary pressure time histories are produced.

Table A3.8-2 indicates the maximum and minimum values for the zonal Fourier coefficients (A_n, B_n) for the first four harmonics.

Table A3.8-3 summarizes the minimum and maximum point (local) pressures calculated for each zone.

Figures A3.8-5 through A3.8-9 show representative values of the normalized Fourier cosine coefficients (a_n) as they vary in time throughout the transient.

A3.8.2.2.6.2 Asymmetric Discharge Case

The asymmetric discharge case is defined as the subsequent actuation of the low setpoint valve together with the simultaneous failure (open) of the valve connected to an adjacent discharge line. The bubble dynamics are produced as discussed above, with the exception that the discharge line undergoing the subsequent discharge is considered to have an elevated water level due to the assumption that the previous transient has not died out. The loading functions are otherwise prepared as discussed above.

Figure A3.8-3 shows the distribution of the discharge devices in the suppression pool, and the zonal definitions are shown in Figure A3.8-4 as before.

Table A3.8-4 contains the zonal extrema for the Fourier coefficients (A_n, B_n), and Table A3.8-5 shows the zonal extrema for the calculated point (local) pressures.

Figures A3.8-10 through A3.8-14 show representative values of the normalized Fourier cosine coefficients (a_n), indicating their variance with time throughout the analysis.

A3.8.2.3 Quencher Design Loads on Suppression Pool Boundaries

The methodology and procedure for determining design boundary loads with a quencher discharge device are discussed in Section 2.2 of Attachment A3.9.

A3.8.3 DEVELOPMENT OF LOCA LOADS

During a LOCA, the structures forming the containment system and other structures within the containment experience dynamic phenomena. This section provides numerical information on the dynamic loads that these phenomena impose on the CPS containment system structures.

A3.8.3.1 Description of LOCA (DBA)

Figure A3.8-15 shows the events occurring during DBA and the potential loading conditions associated with these events (Reference 18).

If the hypothetical guillotine break is postulated to occur within the annulus between the RPV and the biological shield wall the shield annulus will experience a rapid pressurization. During this short-lived transient the annulus is pressurized before the break flow can eventually find its way into the drywell through the various penetrations and openings in the shield wall and the opening at the top of the shield wall.

As the drywell pressure increases, the water initially standing in the vent system discharges into the pool, clearing the vents. During this vent-clearing process, the water leaving the horizontal vents forms jets in the suppression pool and causes water jet impingement loads on the structures within the suppression pool and on the containment wall opposite the vents.

During the vent-clearing transient, the drywell is subjected to a pressure differential, and the weir wall experiences a vent-clearing reaction force.

Immediately following vent clearing, an air and steam bubble forms at the exit of the vents. The bubble pressure initially is conservatively assumed equal to the current drywell pressure (drywell pressure at vent clearing is calculated as 17.56 psig, however, the conservative value of 20.1 psig is the design basis peak pressure), which transmits a pressure wave through the suppression pool and results in loading on the suppression pool boundaries and on equipment located in the suppression pool.

As air and steam flow from the drywell becomes established in the vent system, the initial vent exit bubble expands to suppression pool hydrostatic pressure. Tests at the GE Large Scale Pressure Suppression Test Facility (PSTF) (Reference 17) show that the steam fraction of the flow is condensed, but that continued injection of drywell air and expansion of the air bubble results in a rise in the suppression pool surface. During the early stages of this process, the pool swells in a bulk mode (i.e., a slug of solid water is accelerated upward by the air). During this phase of pool swell, structures close to the pool surface will experience impact loads as the rising pool surface reaches the lower surface of the structure. In addition to these initial impact loads, these same structures will experience drag loads as water flows past them. Equipment in the suppression pool located above the elevation of the bottom vent will also experience drag loads.

Data from PSTF air tests indicates that after the pool surface has risen approximately 1.6 times the initial submergence of the top vent, the water ligament thickness has decreased to 2 feet or less when the ligament begins to break up, and the impact loads are significantly reduced. This phase is referred to as incipient breakthrough.

Ligament thickness continues to decrease until complete breakthrough is reached and the air bubble can vent to the containment free space. The breakthrough process results in formation of an air/water froth.

Continued injection of drywell air into the suppression pool results in froth pool swell. This froth swell impinges on structures it encounters, but the two-phase nature of the fluid results in loads that are much smaller than the impact loads associated with bulk pool swell.

When the froth reaches the bottom elevation of the open portion of the floors on which the hydraulic control units for the control rod drives are located, approximately 22 feet above pool level, the froth encounters a flow restriction; at this elevation, there is approximately 35% open area. The froth pool swell experiences a two-phase pressure drop as it flows through the available open areas. This pressure differential represents a load on both the floor structures and the adjacent containment and drywell. The result is a discontinuous pressure loading at this elevation.

The pool swell impact and impingement target data presented in Attachment A3.9 are applicable to small structures (less than 20 inches). This restriction on the application of the impact test data is necessary, since the basic tests involved targets with a width of 20 inches. For this size target, only the suppression pool water in the immediate vicinity of the target has to be redirected by the impact impulse, thus the impact loads are not dependent upon the pool swell water ligament thickness. Attachment A3.9 discusses application of PSTF impact data to small structures.

For floors that are expansive enough to decelerate a large sector of the pool rather than a small region of the pool in the vicinity of the target, the impulsive loading on the floor is dependent upon the momentum of the entire slug and is related to slug thickness.

As drywell air flow through the horizontal vent system decreases and the air/water suppression pool mixture experiences gravity-induced phase separation, pool upward movement stops, and the "fallback" process starts. During this process, floors and other flat structures experience downward loading. No pressure increase on the containment wall has been observed experimentally.

The pool swell transient associated with drywell air venting to the pool typically lasts 3 to 5 seconds. Following this, there is a long period of high steam flow rate through the vent system; data indicates that this steam will be entirely condensed in a region adjacent to the vent exits. For the DBA reactor blowdown, steam condensation lasts for a period of approximately 1 minute. Structural loadings during the steam condensation phase of the accident are relatively small and are included in the containment loading specification.

As the reactor blowdown proceeds, the primary system is depleted of high-energy fluid inventory, and the steam flow rate to the vent system decreases. This reduced steam flow rate leads to a reduction in the drywell/containment pressure differential, which in turn results in a sequential re-covering of the horizontal vents. Re-covering of a particular vent row occurs when the vent stagnation differential pressure corresponds to the suppression pool hydrostatic pressure at the row of vents.

Toward the end of the reactor blowdown, the top row of vents is capable of condensing the reduced blowdown flow, and the two lower rows will be totally re-covered. As the blowdown steam flow decreases to very low values, the water in the top row of vents starts to oscillate back and froth, causing what has become known as vent "chugging." This action results in dynamic loads on the top vents and on the weir wall opposite the upper row of vents and on the drywell and containment. Since this phenomenon is steam mass flux dependent (the chugging threshold appears to be in the range of 10 $lb/sec/ft^2$), it is present for all break sizes. For smaller breaks, it is the only mode of condensation that the vent system will experience.

Shortly after a DBA, the emergency core cooling system (ECCS) pumps automatically start up and pump condensate and/or suppression pool water into the reactor pressure vessel. This water cascades into the drywell from the break. The time at which this occurs depends upon break size and location. Because the drywell is full of steam at the time of vessel flooding, the sudden introduction of cool water causes steam condensation and drywell depressurization. When the drywell pressure falls below the containment pressure, the drywell vacuum relief valve open and air from the containment enters the drywell to equalize the drywell and containment pressure. During this drywell depressurization transient, there is a period of negative pressure on the drywell structure, and a conservative negative load condition is therefore specified for the drywell design. While the drywell pressure is below the containment pressure the containment atmosphere flows from the wetwell through the vacuum breakers to the drywell and the suppression pool water flows through the LOCA vents into the drywell. The suppression pool water spills over the weir wall and induces drag loads on piping, equipment, and structures below the top of the weir wall. Impact and drag velocities are computed by methods similar to those described in Reference 21. Impact and drag loads resulting from the upward movement of the water in the weir annulus are computed by the methods described in Appendix L of Reference 21. The downward drag loads due to weir swell are computed by the same method as the pool swell fall back loads with the velocity being computed as the free fall velocity from the maximum calculated weir swell height (740'-9").

Suppression pool water is continuously recirculated through the core by the ECCS pumps, resulting in a slow heatup of the suppression pool. To control suppression pool temperature, the RHR system is put into service. After several hours, the heat exchangers control and limit the suppression pool temperature increase. It is conservatively calculated that the pool will reach a peak temperature of 182.2°F. The increase in air and water vapor pressure at these temperatures results in a pressure loading of the containment, as discussed in Chapter 6.

The post-DBA containment heatup and pressurization transient is terminated when the RHR heat exchangers reduce the pool temperature and containment pressure to nominal values.

A3.8.3.2 Description of LOCA (IBA)

An intermediate size break is defined as a break that is less than the DBA but is of sufficient magnitude to automatically depressurize the primary system due to loss of fluid and/or automatic initiation of the ECCS systems (Reference 18). In practice, this means liquid breaks greater than 0.05 ft^2 and steam breaks greater than 0.4 ft^2 as determined by analysis.

In general, the magnitude of dynamic loading conditions associated with a loss-of-coolant accident decreases with decreasing break size. However, the intermediate break is examined because the automatic depressurization system (ADS) is involved and simultaneous actuation of the multiple SRV's committed to this system introduces containment system loads, as discussed in Section A3.8.6.

A3.8.3.3 Description of LOCA (SBA)

Small breaks are defined as breaks not large enough to depressurize the reactor automatically (Reference 18). The dynamic loads produced by this class of accident are small. However, there are certain conditions associated with smaller reactor system breaks that must be considered during the design process. Specifically, the drywell and weir wall must be designed for the thermal loading conditions that can be generated by a small steam break (SBA). For a definition of the design conditions, the following sequence of events is postulated.

With the reactor and containment operating at maximum normal conditions, a small break is postulated to occur, allowing blowdown of reactor steam to the drywell. The resulting drywell pressure increase leads to a high drywell pressure signal that scrams the reactor and activates the containment isolation system. Drywell pressure continues to increase at a rate dependent on the size of the assumed steam leak. A pressure increase to approximately 3 psig depresses the water level in the weir annulus until the level reaches the top of the upper row of vents. At this time, air and steam enter the suppression pool. Steam is condensed, and the air passes to the containment free space. The latter results in gradual pressurization of the containment at a rate dependent upon the air carryover. Entrainment of the drywell air in the steam flow through the vents results in all the drywell air being carried over to the containment. At this time, containment pressurization ceases. The drywell is now full of steam and has a positive pressure differential sufficient to keep the weir annulus water level depressed to the top vents and chugging can occur. Reactor blowdown steam continues to be condensed in the suppression pool.

The thermodynamic process associated with blowdown of primary system fluid is one of constant enthalpy. If the primary system break is below the normal RPV water level, blowdown flow consists of reactor water. Upon depressurizing from reactor pressure to drywell pressure, approximately one-third of this water flashes to steam, two-thirds remains as liquid, and both phases will be in a saturated condition at drywell pressure. Thus, if the drywell is at atmospheric pressure, the steam-and-liquid blowdown will have a temperature of 212° F.

If the primary system rupture is located so that the blowdown flow is reactor steam, the resultant temperature in the drywell is significantly higher than the saturated temperature associated with liquid blowdown. This is because a constant enthalpy decompression of high-pressure saturated steam results in a superheat condition. For example, decompression of 1,000-psia saturated steam to atmospheric pressure results in 298° F superheated steam (86° F of superheat).

Reactor operators are alerted to the SBA incident by the leak detection system, or high drywell-pressure signal, and reactor scram. For the purpose of evaluating the duration of the superheat condition in the drywell, it is assumed that operator response to the small break is to depressurize the reactor using selected relief valves and to activate the RHR system to control the suppression pool temperature. (This conservatively assumes that the main condenser is not available and the operators must use the suppression pool for an energy sink.) Reactor cooldown is assumed to be started 30 minutes after the break and to proceed at a rate of 100° F/hr. This results in a reactor cooldown to 212° F or less in approximately 3 to 6 hours, after which time the blowdown flow rate is terminated. It should be noted that the end-of-blowdown chugging phenomenon discussed in Subsection A3.8.3.4 also will occur during a small-break accident and will last the duration of reactor depressurization.

A3.8.3.4 Design Loads for LOCA Events

The design pressures and temperatures for the drywell which result from the various LOCA phenomena and assumptions are discussed in detail in Subsection 6.2.1.1.3. These analyses are not repeated here, but the consequences of the transients are incorporated in the design as discussed in Subsection A3.8.5, Attachments B3.8 and A3.9, and Sections 3.9 and 3.10. Also, the effects of a pipe break within the shield annulus are discussed in detail in Subsection A3.8.6.2 and accounted for in the design.

The effects of pool swell impact and drag loads on piping and equipment located in the containment are quantified and incorporated into the design as discussed in Attachment A3.9.

The discussions that follow quantify the effects of the LOCA transient that act directly on the containment structure. The incorporation of these effects into the structural design is discussed in Sections A3.8.5 and A3.8.6.

A3.8.3.4.1 Water Jet Loads

During the vent clearing transient, the weir wall, the weir side of the drywell wall and the LOCA vents will experience drag loads due to the suppression pool water being forced through the weir annulus and the LOCA vents by the rising drywell atmospheric pressure. These loads are negligible when compared with other loads (i.e., condensation oscillation and chugging) that act on these surfaces later in the transient.

The water jets emitted from the LOCA vents during the vent-clearing transient impinge on the containment wall. However, the loading imposed on the containment wall is insignificant when compared with the load caused by the air bubble loads that apply later in the transient, as discussed in Reference 18.

A3.8.3.4.2 LOCA Air Bubble Loads

A3.8.3.4.2.1 LOCA Air Bubble Loads on the Weir Annulus and Vent System

Once the flow of air, steam and water droplets has been established in the vent system, there will be a static pressure reduction in the weir annulus and vent system that leads to a uniform outward pressure of approximately 10 psid on the weir wall. This loading was calculated with the flow model described in Reference 15, and for design purposes is assumed to exist for the first 30 seconds of the blowdown (Reference 18).

A3.8.3.4.2.2 LOCA Air Bubble Loads on the Exterior of the Drywell Wall

During the vent clearing process, the drywell pressure reaches a peak pressure of 20.1 psig. During the subsequent vent flow phase of the blowdown, the peak pressure differential does not exceed the 20.1-psid value even when it is assumed that pool swell results in some two-phase flow reaching the containment annulus restriction at the HCU floor. Interaction between the pool swell water mass and the limited number of structures at or near the pool surface does not significantly affect the drywell pressure.

During bubble formation, the outside of the drywell wall will be subject to varying pressures. A bounding range of 0 to 20.1 psid is specified on those sections of the drywell wall below the suppression pool surface. The basis for this specification is the knowledge that the minimum pressure increase is 0 psi, and that the maximum bubble pressure can never exceed the peak drywell pressure of 20.1 psid, as discussed in Reference 18. As the water rises to a maximum of 18 feet above the nominal pool surface, the pressure decreases linerally from 20.1 psid to 0 psid, as shown in Figures A3.8-16 and A3.8-17.

A3.8.3.4.2.3 LOCA Air Bubble Loads on the Containment

The magnitude of the pressure increase on the containment wall following vent clearing is dependent upon the rate at which the drywell air bubble accelerates the suppression pool water. Circumferential variations in the air flow rate may occur due to drywell air/steam mixture variations, but these variations result in negligible variations in the containment bubble pressure load.

The large-scale PSTF test data are the basis for specifying the maximum asymmetic load of 10 psid. A maximum increase of 10 psid on the containment wall was observed in the PSTF at the Mark III drywell peak calculated pressure of 36.5 psia. Thus, use of a 10-psid asymmetric pressure condition applied in a worst-case distribution is a bounding specification for the symmetric load on the containment wall, as discussed in Reference 18.

The symmetric loading specification is based on the same test results and is specified as a symmetric pressure load of 10 psid applied over the containment surface below the suppression pool surface (Reference 18). Above the suppression pool surface, the pressure is attenuated linearly to 0 psid over 18 feet, as shown in Figure A3.8-16. The time history for both the symmetric and asymmetric loading function if given in Figure A3.8-17.

A3.8.3.4.2.4 LOCA Air Bubble Loads on the Basemat

The outer half of the basemat will experience a 10-psid bulk pressure load associated with the air bubble formation discussed above. The inner half of the basemat will experience a pressure rise that is assumed to increase from 10 psid at the center of the pool to 20.1 psid at the drywell wall. The increase in pressure over the inner 50% of the pool width is assumed to be linear, as discussed in Reference 18 and as shown in Figure A3.8-16. The time history for the transient is shown in Figure A3.8-17.

A3.8.3.4.3 Pool Swell Drag and Impact Loads

The loads imposed on the structure and equipment within the pool swell zone are discussed in Attachment A3.9.

For pool swell loading, Clinton piping systems above the suppression pool are typically restrained in the vertical direction utilizing pinned-pinned rigid supports. The concern is related to lateral instability of piping systems under steady-state pool swell/froth drag loading. A review was made of all affected piping systems with significant runs of horizontal piping. This review has shown that the affected piping is rigidly restrained in the lateral direction of seismic/pool dynamic response spectrum loading. The lateral rigid restraints are adequately designed and spaced to eliminate potential instability problems. The NRC concern has therefore been considered in the Clinton design. (Q&R 210.03)

A3.8.3.4.4 Fallback Loads

The drag loads due to pool swell fallback on the drywell, containment and basemat are negligible compared with the initial pool swell drag and bubble loads. Fallback drag loads on piping and equipment are specified in Attachment A3.9.

A3.8.3.4.5 Froth Impingement Loads on the HCU Floor

The HCU floor is approximately 23.5 feet above the suppression pool and is approximately 11 feet above the elevation at which breakthrough is expected to occur. Froth will reach the HCU floor approximately 0.5 second after the top vent clears and will generate both impingement loads on the structure and a load due to flow pressure differential as it passes through the restricted annulus area at this elevation.

This will result in vertical loads on the drywell wall, an upward loading on the HCU floor and an outward pressure loading on the containment wall. The time history for the froth impingement impact and drag loads is shown in Figure A3.8-18. The froth impingement load definition can be determined from Figure A3.8-18a. The HCU floor has been assessed for this load.

Following the completion of the design of the structures and components at the HCU floor elevation, Reference 22 was received which gave the NRC acceptance criteria for the froth impact load. The design of the Clinton Power Station is judged to be adequate for these new loads.

A3.8.3.4.6 Condensation Oscillation Loads

Following the initial LOCA pool swell transient, during which the drywell air is vented to the containment free space, there will exist a high steam mass flow through the top vents, and condensation oscillation will occur. Vent mass fluxes up to 25 lb/sec-ft^2 will occur as the result of a main steam or recirculation line break. For a more detailed discussion and for a discussion of the test data on which the following methodology is based, see Reference 18.

The condensation oscillation forcing function used for design is defined as a summation of four harmonically related sine waves developed from a regression analysis of the data obtained in test series 5807 of Reference 16.

The results are:

$$P(\tau) = \frac{A(t)}{2} \{ 0.80 \sin [2\pi\tau\, f(t)] + 0.30 \sin [4\pi\tau\, f(t)] + 0.15 \sin [6\pi\tau\, f(t)] + 0.20 \sin [8\pi\tau\, f(t)] \} \qquad \text{(A3.8-5)}$$

where:

$P(\tau)$ = pressure amplitude for a cycle beginning at time t and ending at t+Tp (psid),

$A(t)$ = peak to peak amplitude variation with time (psid)

$= 5.5[3.395 - 0.106t + 1.15 \log t - 7.987(\log t)^2 + 7.688 (\log t)^3 - 1.344 (\log t)^4,$ (A3.8-6)

$f(t)$ = fundamental frequency variation with time (sec-:)

$= 0.8[2.495-0.225t - 0.742 \log t + 10.514(\log t)^2 - 9.271(\log t)^3 + 3.208(\log t)^4],$ (A3.8-7)

t = time (sec) from initiation of LOCA, $3 \le t \le 30$,

= time (sec) from the beginning of each cycle, $0 \le \tau \le Tp$, and

Tp = $f(t)^{-1}$ (sec).

$P(\tau)$ from Equation A3.8-5 has been calculated for four cycles and is shown in Figure A3.8-19. The spatial distribution of the forcing function amplitude over the wetted surface of the suppression pool is shown in Figure A3.8-20. The amplitudes shown in Figure A3.8-20 are normalized to 1.0 at the centerline of the top vent.

A3.8.3.4.7 Chugging Loads

In addition to the bulk drywell pressure fluctuations, pressure pulses have been observed when steam bubbles collapse in the vents, in a process known as chugging. The dominant pressure response to the top vent during chugging is that of a pulse train, with the peak amplitude of the pulses varying randomly from chug to chug. The pulse train associated with a chug consists of a sequence of four pulses with exponentially decreasing amplitude.

The dominant pressure response to chugging in the suppression pool is characterized by a pre-chug underpressure, chug (pressure spike), and a post-chug oscillation.

The chugging phenomenon as observed in PSTF tests has a random amplitude and frequency. Although it is expected that chugging will occur randomly among the vents, synchronous chugging in all top vents is conservatively assumed. Each vent is expected to be periodically subjected to the peak observed pressure spike. The pool boundary load definitions consider that the chugging loads transmitted to the drywell wall, base mat, weir wall, and containment are the result of several vents chugging simultaneously at different amplitudes.

A detailed discussion of the data on which the following load definitions are based is presented in Reference 18.

A3.8.3.4.7.1 Chugging Loads Applied to Top Vent

Within the top vent, the peak pressure pulse train shown in Figure A3.8-21 is applied for local or independent evaluation of vents. Although some variation is observed in the pressure distribution from the top to the bottom of the vent, it is conservatively assumed that during the chugging event the entire top vent wall is simultaneously exposed to spatially uniform pressure pulses. Because some net imbalance in the pressure distribution gives rise to a vertical load, the peak force pulse train shown in Figure A3.8-22 is applied vertically upward over the projected vent area concurrently with the peak pressure pulse train shown in Figure A3.8-21 to evaluate local effects on the vents. For global effect, the average force pulse train shown in Figure A3.8-23 is applied vertically over the projected area of all top vents concurrently with the average pressure pulse train within the vent shown in Figure A3.8-24.

The underpressure preceding the pressure pulse train within the top vent is very small compared with the peak (spike) over-pressure. The mean measured pressure in the vent (results from tests) was -9 psid with a standard deviation of $\pm$ 3 psid. On this basis, the specified design value is -15 psid. A bounding underpressure of -19.5 psid was calculated for inside vent surface.

The pressure pulses generated inside the top vents during chugging propagate toward the weir annulus. The dominant pressure response in the weir annulus during chugging is characterized by a pre-chug underpressure followed by a pressure pulse train as shown in Figure A3.8-25. The load applied to the weir annulus (weir wall and drywell wall) is described by a prechug underpressure, defined as a half sine wave as shown in Figure A3.8-26, followed by a pressure pulse train as shown in Figures A3.8-27 and A3.8-28. For local load considerations, the peak amplitudes are applied. For global considerations, the mean amplitudes are applied.

Vertical attenuation of the weir underpressure is small; for design evaluation, no attenuation is assumed. For the pressure pulse train, the attenuation on the weir wall and the drywell ID wall in the vertical direction is shown in Figure A3.8-29. For distribution in the circumferential direction, the local loading for the pre-chug underpressure attenuates as shown in Figure A3.8-29. For the global loads, there is no attenuation in the circumferential direction.

A3.8.3.4.7.2 Pool Boundary Chugging Loads

The chugging load applied to the pool boundary (drywell, base mat and containment) is described by the typical forcing function shown in Figure A3.8-30. The forcing function consists of a prechug underpressure defined as a half sine wave, a triangular pulse (pressure spike) loading characterized by a time duration "d" and a post-chug oscillation described by a damped sinusoid. The impulse is at its maximum magnitude and duration near the top vent on the drywell wall due to the localized nature of the phenomena. The amplitude of the pre-chug underpressure and the post-chug oscillation are also maximum at this location.

For local load considerations on the pool boundary:

Pre-chug underpressure:

peak amplitude - Table A3.8-6; and
distribution - Figure A3.8-31.

Pulse (spike):

peak amplitude - Table A3.8-6;
distribution - Figure A3.8-32; and
duration - Figure A3.8-33.

Post-chug oscillation:

peak amplitude - Table A3.8-6; and
distribution - Figure A3.8-34:

For distribution in the horizontal (circumferential) direction, the pre-chug underpressure attenuates on the drywell, base mat and containment, as shown in Figure A3.8-35. The pulse attenuation is the same as the lower portion of the vertical attenuation shown in Figure A3.8-32, except that the peak is at the vent centerline, and the post-chug oscillation attenuates on the drywell, base mat and containment, as shown in Figure A3.8-36. The profiles in Figures A3.8-35 and A3.8-36 represent the peak observed value at one vent, with the other vents chugging at the mean value.

For global load considerations on the pool boundary:

Pre-chug underpressure:

mean amplitude - Table A3.8-6; and
distribution - Figure A3.8-37a.

Pulse (spike):

mean amplitude - Table A3.8-6;
distribution - Figure A3.8-32; and
duration - Figure A3.8-33.

Post-chug oscillation:

mean amplitude - Table A3.8-6; and
distribution - Figure A3.8-37.

There is no horizontal attenuation for this loading.

A3.8.3.4.7.3 Cyclic Temperature Due to Chugging

The chugging phenomenon includes the continual evacuation and reflooding of the vents and periodic collapse of steam bubbles on the drywell wall. The temperature transient and area of application are shown in Figures A3.8-38 and A3.8-39.

A3.8.3.4.7.4 Suppression Pool Thermal Stratification

During the period of steam condensation in the suppression pool, the pool water in the immediate vicinity of the vents is heated. For the Mark III configuration, most of the condensing steam mass and energy are released to the pool through the top vents. The hot water rises by natural convection, and the cold water is displaced toward the bottom of the pool. The vertical temperature gradient resulting from this effect is known as thermal stratification. The short-term thermal stratification for the large break accident used in the containment evaluation is shown in Figure A3.8-40.

A3.8.4 DEFINITION OF OTHER LOADS

Loads other than the SRV and LOCA loads used in the structural design of the containment structure are discussed in Chapter 3 of the FSAR. Variation of suppression pool temperature with depth is accounted for in the design of the containment.

A3.8.5 LOAD COMBINATIONS

Load combinations used in the structural design of the containment structure are discussed in Section 3.8.

A3.8.6 ANALYSIS METHODS

A3.8.6.1 General

Dynamic structural analysis for the SRV and LOCA loads is discussed in Subsections A3.8.6.1.1 and A3.8.6.2. A brief description of each of the loads precedes the discussion of the analysis.

Structural responses to pool dynamic loads are most pronounced in the region where the loads occur, namely, the suppression pool. The suppression pool region is bounded by axisymmetric structures such as the containment wall, drywell wall, and the basemat. Therefore, it is appropriate and efficient to use axisymmetric thin shell of revolution structural responses can be accurately determined.

The presence of adjacent structures is accounted for by including in the analysis model the adjacent structures that will have significant response to pool dynamic loads. These structures are represented by axisymmetric plates and shells having masses and stiffnesses equivalent to their actual configuration. The formulation of these axisymmetric finite elements does not deviate from the general finite element approach.

Unlike the seismic phenomenon where the loading represents ground motion, the pool dynamic loads represent conditions of forced vibration which is relatively localized and the structural responses attenuate away with distance from the source of the loading. In addition, the free boundary conditions at the outer edge of the modeled base slab conservatively ignore any stiffening effect of the base slab extension to the other unit. Therefore, it is not necessary to model the two unit complex. (Q&R 220.43)

Typical response of the structure is shown in Figures A3.8-41 through A3.8-46.

A3.8.6.1.1 Analysis of SRV Loading

SRV loads for both the rams head and quencher discharge devices have been considered in the CPS structural analysis. The preliminary design of the containment was based on the conservative rams head load. It was later decided that the quencher device would be installed at CPS, since it imposed a reduced loading. The containment structure originally designed for the rams head loads also is assessed for the quencher loads.

The analysis for the rams head load is presented in Section A3.8.6.1.1.1.

The analysis for the quencher load is described in Section A3.8.6.1.1.2.

A3.8.6.1.1.1 Rams Head Discharge

A3.8.6.1.1.1.1 Loading

The rams head SRV loads are described in detail in Section A3.8.2. Both a symmetric and an asymmetric load case are considered. The peak positive pressure for the symmetric rams head load is 83.5 psid and the peak negative pressure is 21.6 psid. The peak positive and negative pressures for the asymmetric load case are 70.0 psid and 11.5 psid, respectively. The frequency of the loads varies approximately from 5 hertz to 13 hertz, and the load has a time duration of approximately 0.75 seconds.

A3.8.6.1.1.1.2 Model for Analysis

The plant structure and underlying soil, were analyzed for the rams head discharge using an axisymmetric finite element model. The soil-structure model is similar to the model used for the quencher analysis shown in Figure A3.8-48. Appropriate material properties are used to describe the various structural and soil model components. The structural components were represented by thinshell finite elements. Solid finite elements were used to model the soil from bedrock up to the basemat.

To consider soil effects beyond the finite element model boundary, an artificial viscous boundary (nonreflecting boundary) is used for horizontal motion. The viscous boundary provides the capability to model an infinitely long system utilizing a model of finite size. The vertical motion at the side boundary of the modeled soil media is assumed to be unrestrained. At the bottom interface between silt and bedrock, fixed boundary conditions are used. More information concerning the viscous boundary can be found in Appendix C - DYNAX. (Q&R 220.42)

Soil properties are strain dependent and are given in Table 2.5-48. In order to obtain appropriate elastic modulus corresponding to the strain, an iterative procedure utilizing an axisymmetric DYNAX model is carried out. Iteration is continued until the assumed strain levels and the actual strain level converge.

The final values of strains and the corresponding soil moduli used in the analysis are shown in Table 3.8-7. (Q&R 220.41)

A3.8.6.1.1.1.3 Method of Analysis

DYNAX, a validated Sargent & Lundy (S&L) proprietary computer program, was used in the dynamic analysis of the rams head loading. DYNAX is a finite element program for the static or dynamic analysis of axisymmetric shells and solids. A description of the program along with validation information is presented in Appendix C. The analysis for the rams head loading was performed using white noise transfer functions. Structural responses were calculated in the frequency domain and then transferred to the time domain using the S&L proprietary computer program FAST. A description of the program along with validation information is presented in Appendix C.

A3.8.6.1.1.2 Quencher Discharge

A3.8.6.1.1.2.1 Loading

The quencher SRV loadings are described in detail in Section A3.9.2.2 of Attachment A3.9. Fifty-nine trial load cases for the quencher discharge were supplied by GE. Out of the 59 load cases, 11 were randomly selected for analysis to represent a nonexceedance probability of at least 84/84.

Three cases of quencher SRV actuations were considered in this analysis: All Valve, automatic depressurization system (ADS), and single valve - subsequent actuation.

A3.8.6.1.1.2.2 Model for Analysis

The analysis for the quencher discharge loading was done using an axisymmetric model. Figure A3.8-48 shows the soil-structure model. Thin-shell finite elements were used to model the structures. The axisymmetric thin-shell finite element models of the RPV are supplied by the General Electric Company. The soil is modeled using solid elements. Fluid finite elements described in Reference 19 are used to simulate the suppression pool water.

A3.8.6.1.1.2.3 Method of Analysis

The DYNAX computer program was used in the analysis for the quencher loads. The analysis was performed using white noise transfer functions and the FAST Fourier Transform of a single bubble loading (Reference 20). Responses for the multiple bubbles were obtained by appropriate superposition of various single bubble responses.

The RSG computer program was used to obtain a response spectrum from the acceleration time history for each of the 11 randomly selected load cases. The response spectra for equipment and subsystem design were then obtained by enveloping the response spectra for the 11 loadings. RSG, a validated S&L proprietary computer program, is described in Appendix C.

A3.8.6.2 <u>Analysis for LOCA Loads</u>

The LOCA loads included for structural analysis were those due to annulus pressurization, LOCA bubble, froth impingement, condensation oscillation, and chugging. Descriptions of the analysis methods for each of the loads are presented in Subsections A3.8.6.2.1 through A3.8.6.2.5, respectively.

A3.8.6.2.1 <u>Annulus Pressurization</u>

Annulus pressurization loads were considered in the shield wall analysis and design. A discussion regarding annulus pressurization can be found in Subsection 6.2.1.2.3.2.

A3.8.6.2.2 <u>LOCA Bubble</u>

A3.8.6.2.2.1 <u>Loading</u>

The LOCA bubble loading is described in detail in Section A3.8.3 of this attachment.

A3.8.6.2.2.2 <u>Model for Analysis</u>

The finite-element model described in Subsection A3.8.6.1.1.2.2 (Figure A3.8-48) was used to analyze the LOCA bubble load effects.

A3.8.6.2.2.3 <u>Method of Analysis</u>

The DYNAX computer program was used to obtain the dynamic structural responses due to the LOCA bubble load time histories. Displacements, accelerations, forces and moments of the structures were obtained using a direct numerical integration algorithm to solve the governing differential equations.

The LOCA bubble pressure loading was applied to the model as equivalent concentrated nodal force time histories. These time histories were applied on the suppression pool basemat ad on the pool walls to a height 18 feet above the pool surface. The nodal force time histories represent the meriodinal distribution of the load on the drywell and containment walls and the radial distribution of the pressure on the basemat. Fourier harmonics were used to account for the circumferential distribution of the load.

The resultant acceleration response time histories at critical locations were used to obtain response spectra for equipment and sub-system design. RSG was used to calculate the acceleration response spectra.

The asymmetric LOCA bubble load was applied over 180° of the suppression pool boundary, including basemat and pool walls. Responses due to this asymmetric loading are considered only for local containment structural evaluation.

A3.8.6.2.3 <u>Froth Impingement</u>

A3.8.6.2.3.1 <u>Loading</u>

The froth impingement loading is described in detail in Section A3.8.3 of this attachment. The peak froth impingement pressure is 15 psid. The load time history is shown in Figure A3.8-18.

A3.8.6.2.3.2 Model for Analysis

The finite-element model described in Subsection A3.8.6.1.1.2.2 (Figure A3.8-48) was used to analyze the froth impingement load effects.

A3.8.6.2.3.3 Method of Analysis

The method of dynamic analysis for the froth impingement load is similar to that described in Subsection A3.8.6.2.2.3. The impingement pressures are applied as nodal load time histories to the drywell floor.

Structural responses are obtained using the DYNAX computer program. Acceleration response spectra are obtained at critical locations for equipment and subsystem design using the RSG computer program.

A3.8.6.2.4 Condensation Oscillation

A3.8.6.2.4.1 Loading

The condensation oscillation loading is described in detail in Section A3.8.3 of this attachment. The peak pressure specified for design is ± 7 psid, as shown in Figure A3.8-19. The frequency of the condensation oscillation load varies, as shown in Equation A3.8-7.

A3.8.6.2.4.2 Model for Analysis

The finite-element model described in Subsection A3.8.6.1.1.2.2 (Figure A3.8-48) was used to analyze the condensation oscillation load effects.

A3.8.6.2.4.3 Method of Analysis

The dynamic analysis for the condensation oscillation load is similar to that described in Subsection A3.8.6.2.2.3. The DYNAX computer program was used to obtain structural responses for the axisymmetric condensation oscillation pressures. The load was applied to the suppression pool boundary as nodal force time histories. The RSG computer program was used to obtain acceleration response spectra for equipment and subsystem design at critical locations.

A3.8.6.2.5 Chugging

A3.8.6.2.5.1 Loading

The chugging loading is described in detail in Section A3.8.3 of this attachment. Symmetric and asymmetric chugging loads are defined by GE. The time histories applied at the top vent on the drywell wall consist of a vertical force pulse train made up of triangularly shaped spikes. The peak pressure pulse train occurs within the top vent and has a magnitude of 540 psid. Refer to Figure A3.8-21.

The time histories applied in the weir annulus area consist of two phases: a sinusoidal pre-chug underpressure, and a pressure pulse train made up of triangularly shaped spikes. The peak pressure pulse train applied to the weir annulus is 43 psid. Refer to Figure A3.8-27.

The time histories applied in the suppression pool area consist of three phases: a sinusoidal pre-chug underpressure, a triangularly shaped pulse (pressure spike), and a post-chug oscillation defined as an exponentially decaying sine function. The peak spike pressure (i.e., the peak time history pressure) occurs on the drywell wall and has a magnitude of 100 psid. Refer to Figure A3.8-30 and Table A3.8-6.

A3.8.6.2.5.2 Model for Analysis

The finite-element model described in Subsection A3.8.6.1.1.2.2. (Figure A3.8-48) was used to analyze the chugging load effects.

A3.8.6.2.5.3 Method of Analysis

The method of analysis for the symmetric and asymmetric chugging load is similar to that described in Subsection A3.8.6.2.2.3 Loads were applied to the containment wall, drywell wall, and base mat as nodal force time histories. Responses due to the symmetric and asymmetric chugging loads were obtained using the DYNAX computer program.

Acceleration response spectra due to symmetric chugging loads were obtained at critical locations for equipment and subsystem design using the RSG computer program. Responses due to asymmetric chugging loads are considered only for local containment structural evaluation.

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TABLE A3.8-1
BUBBLE DYNAMICS EQUATIONS

Differential Equations:

$$\dot{P}P = 3k\left[\frac{n}{4\pi R^3}\frac{P_d}{\rho_d}\dot{m}_b - P\frac{\dot{R}}{R}\right] \tag{3.8A-8}$$

$$\ddot{R} = \left[(P - P\infty)\frac{g_c}{\rho_L} - \frac{3}{2}\quad \dot{R}^{2}\right]/R \tag{3.8A-9}$$

where:

$P\infty = P_{ref} + \rho_L{}^{Z.}$

For (t > td):

$$\ddot{Z} = F/m_v$$

where

$$= -\frac{1}{2}\pi\rho_L\ C_D R^2\ \dot{Z}\left|\dot{Z}\right| - \frac{4}{3}\pi\rho_L g R^3 + m_b g\text{, and} \tag{3.8A-10}$$

$$m_v = m_b + \frac{-2}{b}\pi\rho_L R^3. \tag{3.8A-11}$$

Then $\dot{z}_2 = \dot{z}_1 + \int_{t_1}^{t_2} \ddot{z}_t\, dt$, and (3.8A-12)

TABLE A3.8-1 (Cont'd)

$$z_2 = z_1 + \int_{t_1}^{t_2} \dot{z}_t \, dt - \dot{R}(t_2 - t_1) \text{ * last term included for } \dot{R} > 0 \text{ only} \qquad (3.8A\text{-}13)$$

Initial Conditions (t = 0):

$$P = \frac{P_d}{2} + \frac{\rho_d V_d^2}{4g}; V_d^2 = gk\frac{P_d}{\rho_d}; R = \frac{D}{2}; \dot{R} = \frac{V_L}{8} \qquad (3.8A\text{-}15)$$

$$Z = L_s; \dot{z} = 0(0 \le t \le t_d); \dot{m}_b = \pi R^2 \frac{\rho_d}{2} V_d (- \le t \le t_d) \qquad (2.8A\text{-}16)$$

where:

$$m_b = \pi/8 \ D^2 \ (L - L_s)\rho_a \text{ for a rams head, and} \qquad (3.8A\text{-}17)$$

$$t_d = m_b / \dot{m}_b . \qquad (3.8A\text{-}18)$$

TABLE A3.8-2
EXTREME VALUES OF COEFFICIENTS FOR
SYMMETRIC DISCHARGE CASE

	ZONE	A_0	B_1	A_1	B_2	A_2	B_3	A_3	B_4	A_4
1.	MAX	10.930	2.204	4.111	0.432	2.027	1.571	1.572	0.432	0.541
	MIN	-3.470	-1.671	-3.266	-1.824	-3.408	-1.694	-1.460	-0.942	-0.726
2.	MAX	26.238	3.862	8.157	0.935	3.542	3.719	3.020	0.902	1.247
	MIN	-7.666	-3.879	-5.614	-4.622	-7.892	-4.367	-2.642	-1.456	-1.651
3.	MAX	26.667	3.573	7.930	0.911	3.412	3.399	2.703	0.802	1.061
	MIN	-7.865	-4.037	-5.179	-4.663	-7.580	-4.055	-2.431	-1.124	-1.386
4.	MAX	25.796	3.431	7.645	0.871	3.259	3.205	2.547	0.748	0.985
	MIN	-7.615	-3.906	-4.973	-4.442	-7.237	-3.840	-2.292	-1.033	-1.254
5.	MAX	21.230	2.828	6.227	0.688	2.583	2.446	1.947	0.544	0.722
	MIN	-6.254	-3.156	-4.100	-3.491	-5.759	-2.907	-1.751	-0.773	-0.944
6.	MAX	17.511	2.362	5.072	0.539	2.029	1.823	1.454	0.375	0.505
	MIN	-5.146	-2.545	-3.433	-2.713	-4.514	-2.142	-1.306	-0.558	-0.662
7.	MAX	16.611	2.253	4.820	0.508	1.915	1.711	1.365	0.348	0.471
	MIN	-4.879	-2.405	-3.274	-2.554	-4.261	-2.005	-1.226	-0.525	-0.618
8.	MAX	12.742	1.781	3.763	0.394	1.489	1.349	1.078	0.277	0.379
	MIN	-3.735	-1.836	-2.587	-1.967	-3.314	-1.565	-0.968	-0.435	-0.499
9.	MAX	4.863	0.705	1.465	0.151	0.577	0.526	0.432	0.108	0.150
	MIN	-1.422	-0.696	-1.024	-0.751	-1.279	-0.603	-0.378	-0.183	-0.198
	NORM (psi)	26.667	4.037	8.157	4.663	7.592	4.267	3.020	1.456	1.651
	NORM (ksf)	3.5400	0.5513	1.1746	0.67115	1.1364	0.6144	0.4349	0.2097	6.2377

TABLE A3.8-3
EXTREME CALCULATED PRESSURES FOR
SYMMETRIC DISCHARGE CASE*

ZONE	P-WALL** MAX (psid)	P-WALL** MIN (psid)
1	17.20	-4.752
2	39.42	-9.383
3	36.46	-8.850
4	34.88	-8.483
5	28.91	-6.875
6	22.82	-5.570
7	21.62	-5.276
8	16.66	-4.055
9	6.382	-1.548

* The extreme values do not necessarily occur at the same point in space or time for the several zones.

** This is the pressure due to air bubble only, i.e., it does not include hydrostatic pressure.

TABLE A3.8-4
EXTREME VALUES OF FOURIER COEFFICIENTS FOR
ASYMMETRIC DISCHARGE CASE

	ZONE	A_0	B_1	A_1	B_2	A_2	B_3	A_3	B_4	A_4
1.	MAX	1.004	1.821	0.361	1.377	0.543	0.888	0.517	0.479	0.363
	MIN	-0.436	-0.734	-0.969	-0.422	-1.303	-0.567	-0.991	-0.874	-0.935
2.	MAX	2.405	4.342	0.784	3.241	1.125	2.032	1.003	1.014	0.655
	MIN	-0.962	-1.592	-2.318	-0.876	-3.077	-1.252	-2.270	-1.892	-2.047
3.	MAX	2.441	4.339	0.791	3.104	1.089	1.830	0.915	0.849	0.555
	MIN	-0.985	-1.606	-2.322	-0.847	-2.956	-1.132	-2.052	-1.588	-1.711
4.	MAX	2.361	4.182	0.763	2.962	1.041	1.723	0.863	0.784	0.515
	MIN	-0.953	-1.549	-2.239	-0.810	-2.822	-1.067	-1.933	-1.471	-1.585
5.	MAX	1.944	3.408	0.621	2.350	0.824	1.319	0.659	0.577	0.378
	MIN	-0.783	-1.260	-1.824	-0.641	-2.238	-0.816	-1.479	-1.082	-1.166
6.	MAX	1.604	2.777	0.505	1.850	0.647	0.987	0.491	0.406	0.264
	MIN	-0.645	1.024	1.486	0.503	1.760	0.610	1.105	0.759	0.819
7.	MAX	1.521	2.631	0.478	1.747	0.610	0.927	0.461	0.379	0.247
	MIN	-0.611	-0.970	-1.408	-0.475	-1.662	-0.573	-1.038	-0.709	-0.765
8.	MAX	1.167	2.026	0.367	1.359	0.474	0.734	0.363	0.306	0.199
	MIN	-0.468	-0.745	-1.083	-0.369	-1.292	-0.453	-0.821	-0.572	-0.619
9.	MAX	0.446	0.776	0.140	0.525	0.182	0.287	0.142	0.122	0.079
	MIN	-0.178	-0.285	-0.415	-0.142	-0.499	-0.177	-0.321	-0.228	-0.246
	NORM (psid)	2.441	4.342	2.322	3.241	3.077	2.032	2.270	1.892	2.047
	NORM (ksf)	0.3515	0.6252	0.3344	0.4667	0.4431	0.2926	0.3269	0.2724	0.2948

TABLE A3.8-5
EXTREME CALCULATED PRESSURES FOR
THE ASYMMETRIC DISCHARGE CASE*

ZONE	P-WALL** MAX (psid)	P-WALL** MIN (psid)
1	9.379	-4.202
2	20.390	-7.611
3	7.120	-6.551
4	16.120	-6.158
5	12.580	-4.756
6	9.789	-3.610
7	9.232	-3.398
8	7.214	-2.663
9	2.796	-1.033

* The extreme values do not necessarily occur at the same point in space or time for the several zones.

** This is the pressure due to air bubble only, i.e., it does not include hydrostatic pressure.

TABLE A3.8-6
CHUGGING LOADS*

LOCATION	PRE-CHUG UNDERPRESSURE AND DURATION		PULSE (SPIKE) AND DURATION "d"		POST-CHUG OSCILLATION AND FREQUENCY	
	PEAK (A)	MEAN (A)	PEAK	MEAN	PEAK (B)	MEAN (B)
Drywell Wall	-5.8 psid 125 ms	-2.65 psid 125 ms	100 psid 8 ms	24 psid 8 ms	±6.50 psid 10-12 hertz	±2.2 psid 10-12 hertz
Containment	-1.3 psid 125 ms	-1.0 psid 125 ms	3 psid 2 ms	0.7 psid 2 ms	±1.7 psid 10-12 hertz	±1.00 psid 10-12 hertz
Base Mat	-1.8 to -1.3 psid 125 ms	-1.34 to -1.0 psid 125 ms	10 to 3 psid 4 to 2 ms	2.4 to 0.7 psid 4 to 2 ms	±2.1 to ±1.7 psid 10-12 hertz	±1.29 to ±1.0 psid 10-12 hertz

* Source: Reference 18

TABLE A3.8-7
(Q&R 220.41)
SOIL STRAIN VERSUS MODULUS

SOIL LAYER* FROM TOP	STRAIN	MODULUS OF ELASTICITY E (ksf)
1	0.1034×10^{-3}	62.5×10^{2}
2	0.1106×10^{-3}	62.5×10^{2}
3	1.1134×10^{-3}	62.5×10^{2}
4	0.1410×10^{-3}	62.5×10^{2}
5	0.1928×10^{-4}	56.0×10^{3}
6	0.1871×10^{-4}	57.0×10^{3}
7	0.1814×10^{-4}	58.0×10^{3}
8	0.1814×10^{-4}	58.0×10^{3}
9	0.1571×10^{-4}	60.0×10^{3}
10	0.1393×10^{-4}	60.0×10^{3}
11	0.1215×10^{-4}	66.0×10^{3}
12	0.9900×10^{-5}	71.0×10^{3}
13	0.7684×10^{-5}	80.0×10^{3}
14	0.7475×10^{-5}	80.0×10^{3}
15	0.7265×10^{-5}	80.0×10^{3}
16	0.1085×10^{-4}	38.1×10^{3}
17	0.8488×10^{-5}	58.0×10^{3}
18	0.7230×10^{-5}	60.0×10^{3}
19	0.6651×10^{-5}	62.5×10^{3}
20	0.6071×10^{-5}	65.0×10^{3}
21	0.7474×10^{-5}	60.0×10^{3}
Bed Rock		

* For soil layer see Figure A3.8-48.

ATTACHMENT B3.8

CONTAINMENT STRUCTURAL DESIGN ASSESSMENT

ATTACHMENT B3.8 - CONTAINMENT STRUCTURAL DESIGN ASSESSMENT

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CONTAINMENT STRUCTURAL DESIGN ASSESSMENT

B3.8.1 INTRODUCTION

The functions and capabilities of the containment structure, the loads and load combinations used in the containment structural design, and the applicable design codes and regulatory guides have been described in Section 3.8 and Attachment A3.8. Structural responses due to normal, pressurization, and earthquake loads have been included in Section 3.8.

This attachment presents responses of the containment structure due to the safety/relief valve (SRV) and loss-of-coolant accident (LOCA) suppression pool dynamic loads, and an assessment of the containment structure for the final design loads.

Section B3.8.2 provides design responses of the containment wall and dome and the drywell structures due to the SRV and LOCA loads.

Section B3.8.3 includes the assessment of the structural capacity with respect to the critical load combinations for the final design loads.

B3.8.2 STRUCTURAL RESPONSE DUE TO SRV AND LOCA LOADS

Responses of the containment wall and dome and the drywell for loads other than SRV and LOCA have been presented in Section 3.8. Containment wall and dome force responses due to dead load, suppression pool hydrostatic load, accident pressure load, and safe-shutdown earthquake are shown in Figure 3.8-17. Similar responses for the drywell are shown in Figure 3.8-33.

Typical force response plots of the containment wall and dome due to the SRV and LOCA loads are shown in Figures B3.8-1 through B3.8-5. Typical drywell force response plots are presented in Figures B3.8-6 through B3.8-10. It should be noted that these plots represent the envelope of the maximum positive and the envelope of the maximum negative values that occur along the height of the structure and that the positive and negative values at a particular elevation do not occur concurrently. Response plots for the SRV loading are for the all valve case and for reasons of simplicity are plotted for only one of the eleven trials which were analyzed (refer to Subsection A3.8.6.1.1.2). For the LOCA loads, the symmetric loading case is used for the force plots. In the assessment of the structural capacity, however, various trials of the SRV loads and symmetric and asymmetric cases of the LOCA loads are enveloped.

B3.8.3 DESIGN ASSESSMENT

Results of the design assessment for the containment structure are provided in Tables B3.8-1 and B3.8-2. Six representative locations on the basemat, containment wall, containment dome, and drywell are considered as shown in Figure B3.8-11.

Concrete and reinforcing stresses due to membrane and flexural forces are calculated for both meridional and circumferential sections. Sections are evaluated for the loads and load combinations shown in Table 3.8-1.1. The critical stresses for the service and factored load categories are tabulated for two conditions, loads without temperature (Table B3.8-1) and loads with temperature (Table B3.8-2).

The stresses due to combined bending and axial loads with and without temperature loads are calculated using the Sargent and Lundy proprietary computer program TEMCO. The program is described in Appendix C.

B3.8.4 CONCLUSION

As shown in Tables B3.8-1 and B3.8-2, the stresses are all within the allowables for the final design loads and therefore the design is adequate.

TABLE B3.8-1
DESIGN ASSESSMENT STRESSES FOR LOADS WITHOUT TEMPERATURE

	SERVICE CONDITION				FACTORED CONDITION			
	REINFORCING TENSION(ksi)		CONCRETE COMPRESSION(psi)		REINFORCING TENSION(ksi)		CONCRETE COMPRESSION(psi)	
SECTION(1)	MERIDIONAL	HOOP	MERIDIONAL	HOOP	MERIDIONAL	HOOP	MERIDIONAL	HOOP
1	30 (4)	17 (4)	840 (4)	690 (4)	43 (19)	20 (19)	1150 (19)	810 (19)
2	6 (3)	26 (3)	510 (4)	220 (4)	22 (19)	37 (13)	630 (19)	250 (11)
3	19 (3)	23 (3)	160 (4)	140 (4)	27 (6)	32 (6)	190 (11)	170 (11)
4	18 (4)	11 (4)	1260 (4)	570 (4)	46 (19)	23 (19)	1620 (11)	870 (19)
5	19 (4)	24 (3)	740 (4)	370 (3)	43 (26)	42 (25)	1110 (26)	670 (25)
6	30 (4)	21 (4)	650 (4)	710 (4)	36 (19)	30 (26)	950 (19)	920 (19)
Allowable Stresses	30 ksi		1800 psi		54 ksi		3000 psi	

(1) See Figure B3.8-11 for Section locations.

(2) Numbers in parentheses identify the load combinations corresponding to the stress values given. Refer to Table 3.8-1.1 and following note (3) for load combination information.

(3) Load combinations 24, 25, and 26 are load combinations 6, 13, and 19 of Table 3.8-1.1 respectively, except with design pressure of 30 psi and without SRV loads. Load combinations 27 and 28 are load combinations 13 and 19 except with long term temperature and without accident pressure or compartment pressure, no LOCA or SRV loads.

TABLE B3.8-2
DESIGN ASSESSMENT STRESSES FOR LOADS WITH TEMPERATURE

	SERVICE CONDITION				FACTORED CONDITION			
	REINFORCING TENSION(ksi)		CONCRETE COMPRESSION(psi)		REINFORCING TENSION(ksi)		CONCRETE COMPRESSION(psi)	
SECTION[1]	MERIDIONAL	HOOP	MERIDIONAL	HOOP	MERIDIONAL	HOOP	MERIDIONAL	HOOP
1	29 (4)	19 (4)	780 (4)	780 (4)	41 (19)	22 (19)	1090 (19)	890 (19)
2	11 (3)	29 (3)	810 (4)	410 (4)	35 (19)	35 (13)	1250 (19)	630 (19)
3	17 (3)	17 (3)	790 (4)	830 (4)	31 (6)	27 (19)	1130 (19)	1110 (19)
4	19 (4)	8 (4)	1790 (4)	1260 (4)	37 (11)	17 (11)	2110 (11)	1440 (11)
5	30 (4)	28 (4)	1030 (4)	330 (4)	54 (26)	55 (26)	1630 (19)	1310 (25)
6	39 (4)	33 (4)	1110 (4)	1240 (4)	42 (28)	43 (28)	1520 (19)	1580 (19)
Allowable Stresses	40 ksi		1800 psi		60 ksi		3000 psi	

(1) See Figure B3.8-11 for Section locations.

(2) Numbers in parentheses identify the load combinations corresponding to the stress values given. Refer to Table 3.8-1.1 and following note (3) for load combination information.

(3) Load combinations 24, 25, and 26 are load combinations 6, 13, and 19 of Table 3.8-1.1 respectively, except with design pressure of 30 psi and without SRV loads. Load combinations 27 and 28 are load combinations 13 and 19 except with long term temperature and without accident pressure or compartment pressure, no LOCA or SRV loads.

ATTACHMENT C3.8

EVALUATION OF SAFETY-RELATED

MASONRY WALLS

EVALUATION OF SAFETY-RELATED MASONRY WALLS

C3.8.1 Criteria Used for Design of Category 1 Masonry Walls at Clinton Station

The loads and load combinations used for the design of Category 1 masonry walls for the Clinton Station are shown in Table C3.8-1. The material properties for the masonry walls are shown in Table C3.8-2. The basic allowable stresses for unreinforced masonry walls are those given in UBC 1979 for inspected workmanship and are shown in Table C3.8-3. The seismic damping values for masonry walls are 2% for normal/severe load combinations and 4% for extreme load combinations.

C3.8.2 Design and Analysis Considerations

The seismic analysis for the safety related concrete masonry walls is in accordance with the requirements of the Clinton Station USAR, Section 3.7. The assumptions and modeling techniques consider proper boundary conditions, cracking of concrete, if any, and the dynamic behavior of the masonry walls. The analysis of the safety related masonry walls consider both in-plane and out-of-plane loads, and interstory drift effects are considered.

There are no concrete masonry shear walls at the Clinton Station, and there are no safety related concrete masonry walls which are subject to accident pipe reaction (Yr), jet impingement (Yj), or missile impact (Ym).

Table C3.8-1
Load Combinations for Category 1 Concrete Masonry

Load Category	D	L	E	E^1	SRV*			LOCA - Pool Dynamics*				Allowable Stresses	
					ALL	1V2P	ADS	PS	CH	CO	MYC	Masonry	Steel
Normal	1.0	1.0											AISC
	1.0				1.0	1.0							
Sovern Environmental	1.0		1.0		1.0	1.0							AISC
Abnormal	1.0					1.0		1.0	1.0	1.0	1.0	(UBC Allowable)	1.6 x AISC $\leq$ .95 Fy
	1.0						1.0		1.0	1.0			
Extreme Environmental	1.0			1.0	1.0	1.0							1.6 x AISC $\leq$ .95 Fy
Abnormal/Severe Environmental	1.0		1.0			1.0		1.0	1.0	1.0	1.0		1.6 x AISC $\leq$ .95 Fy
	1.0		1.0				1.0		1.0	1.0			
Abnormal/Extreme Environmental	1.0			1.0		1.0		1.0	1.0	1.0	1.0		1.6 x AISC $\leq$ .95 Fy
	1.0			1.0			1.0		1.0	1.0			

*Only one load under each of these loadings shall be considered at one time.

Load Symbols are defined as follows:

- D = Dead load of masonry wall including atachment loads
- L = Live load
- E = Operating Basis Earthquake (OBE)
- E^1 = Safe Shutdown Earthquake (SSE)
- SRV_{1V2P} = Safety/Relief Valve (SRV) discharge loading due to discharge of one Safety/Relief Valve subsequent activation.
- SRV_{ADS} = SRV loading due to seven (ADS) Safety/Relief Valve discharges
- SRV_{ALLV} = SRV loadng due to 16 (ALL) Safety/Relief Valve discharges
- LOCA MVC = LOCA loading due to main vent clearing
- LOCA PS = LOCA loading due to pool swell
- LOCA CO = LOCA loading due to condensation oscillation
- LOCA CH = LOCA loading due to chugging

Table C3.8-2
MASONRY MATERIAL PROPERTIES

1. Hollow Concrete Masonry Blocks Minimum ultimate compressive strength (fc) shall be 1000 psi based upon the gross cross-sectional area of the blocks	ASTM C90, Grade N-I
2. Solid Concrete Masonry Blocks Minimum ultimate compressive strength (fc) shall be 1800 psi based upon the gross cross-section area of the blocks.	ASTM C145, Grade N-I
3. Grouted Concrete Masonry Blocks	Hollow Blocks: ASTM C90, N-I Grout : ASTM C476
4. Mortar, Type M*	ASTM C270 with minimum compressive strength m_o = 2,500 psi
5. Masonry Compressive Strength, f_m	1,350 psi, Type M mortar
6. Reinforcement for Concrete Masonry	Truss or ladder type F_y = 65ksi, ASTM A82

*For the Clinton Station Type M mortar shall be used for all masonry construction.

Table C3.8-3
ALLOWABLE STRESSES/STRAINS FOR CONCRETE MASONRY
INSPECTED WORKMANSHIP
UNREINFORCED MASONRY
(Table Values are based on UBC 1979)

			Allowable Stresses (PSI)					
			Normal/Severe Environ. Load Comb.		Extreme Envison. Load Comb			
			Type M Mortar		Type M Mortar		Over Stress Factor Used	Ult Shr Strn
	Type of Stress	Type of Unit (c)	m_o = 2500 PSI		m_o = 2500 PSI			
1.	Compression Flexural and Axial F_m & F_a	H	170		425		2.5	
		S	175		438		2.5	
		G	225		562		2.5	
2.	Bearing Under Conc. Loads	H	255		637		2.5	
		S	262		655		2.5	
		G	337		842		2.5	
3.	Tension in Flexure 1) Normal to bed Joint F_{t1}	H	12[(a)]		12[(a)]		1.0	
		S	12		12		1.0	
		G	12		12		1.0	
	2) Parallel to bed Jonts F_{t11}	H	24		36		1.5	
		S	24		36		1.5	
		G	50		75		1.5	
4.	Shear Out of Plane Loads and In-Plane Loads	H	12		12		1.0	(b) .001
		S	12		12		1.0	
		G	25		25		1.0	

(a) This value is allowed for tensile stresses due to inplane loads for existing walls only. Use F_{t1} = 0 psi for new walls. F_{t1} = 12 psi is also allowed for wall strip around opening spanning vertically.

(b) Shear strain is equal to Δ/h where Δ is relative horiz. floor displacement and h is the height of wall.

(c) H = Hollow Units
G = Grouted Units
S = Solid Units

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

This section shows the transients which are used in the design of the ASME Code Class 1 components. The number of cycles or events for each transient is included. The design transients shown in this section are included in the design specifications for the components. Transients or combinations of transients are classified with respect to the component operating condition categories identified as "normal," "upset," "emergency," "faulted," or "testing" in the ASME Boiler and Pressure Vessel Code, Section III, Subsection NCA. (The first four conditions correspond to Service Levels A, B, C, and D, respectively.)

In the fatigue analysis of NSSS equipment, piping, reactor pressure vessel and RPV internal components, the actual calculated loads due to OBE and SRV are combined to show compliance with upset limits for fatigue. This calculation is performed by comparing the plant unique OBE + SRV load to the original OBE load used for design basis. A stress evaluation is done to show that stresses are within acceptance limits if the plant unique OBE + SRV load exceed the original OBE loads used for the design basis. The larger of the original OBE load or the plant unique OBE + SRV load has been evaluated for 10 or more fatigue cycles consistent with upset limits. For reactor vessel nozzle loads, the original OBE load is also the maximum permissible value shown in the interface control document (ICD) issued by General Electric. Consequently, OBE loads have been combined with other upset loads (including SRV) for the fatigue evaluation.

The number of SRV cycles predicted for a BWR/6 is significantly lower than for a typical BWR/5 because of the low-low set point design. This results in ~13000 SRV cycles for the Clinton BWR/6 reactor system compared to many more (a range of 40 to 80 thousand) cycles predicted for previous lines. (Q&R MEB (DSER) 93) Note: A cycle is one air bubble oscillation in the suppression pool. One SRV opening or actuation results in several cycles.

Bolting used in equipment anchorage is designed to the stress limits given in the AISC Manual, 0.6 Fy for normal and upset loads and 0.95 Fy for emergency and faulted loads for tension allowables. In addition, shear and shear tension interaction are verified.

The bolting used in component supports are of three types: concrete expansion anchors, auxiliary steel bolting, and standard component support bolting.

- The stress limits used for concrete expansion anchor design is provided in IPC's response to I&E Bulletin 79-02. Shell-type anchors are not used in Seismic Category I applications at Clinton Power Station.

- The stress limits used for auxiliary steel bolting is the same as a) above.

- The stress limits used for bolting of standard components is as follows:

Snubber/Strut Loads Studs

Type of Load	Service Levels A and B*	Basis
Tension	0.5 Su	ASME Code Case 1644-6 paragraph 7
Shear	$0.62 \frac{Su}{3}$	ASME Code Case 1644-6 paragraph 7
Bending	0.75 Sy	ASME Section III Appendix XVII paragraph 2214.3
Bearing	1.5 Su	ASME Section III Appendix XVII paragraph 2461.5

* For the Service Level C the allowable stresses are increased by 1.33 per ASME Section III NF 3231.1. For the Service Level D the rules of ASME Section III, Appendix F, paragraph F1370 are used.

Hanger Clamps and Other General Bolting

Tension	0.6 Sy	ASME Section III Appendix XVII paragraph 2211a
Shear	0.4 Sy	ASME Section III Appendix XVII paragraph 2212
Bending	0.75 Sy	ASME Section III Appendix XVII paragraph 2214.3
Bearing	1.35 Sy	ASME Section III Appendix XVII paragraph 2461.2

Load interaction is checked by the method outlined in ASME Section III, Appendix XVII, paragraph 2215.2 (Q&R MED (DSER) 95).

Subsections 3.9.1.1.1 through 3.9.1.1.12 describe the design transients for NSSS systems and components. BOP system and component design transients are described in Subsection 3.9.1.1.13. The cyclic and transient occurrences required to be tracked by the Technical Specifications are identified on Table 3.9-1(b).

3.9.1.1.1 CRD Transients

The normal and test service load cycles used for the design and fatigue analysis for the 40-year life of the control rod drive are as follows:

Transient	Category	
Reactor startup/shutdown	Normal/upset	120
Vessel pressure tests	Normal/upset	130
Vessel overpressure	Normal/upset	10
Scram tests	Normal/upset	140
Startup scrams	Normal/upset	160
Operational scrams	Normal/upset	300
Jog cycles	Normal/upset	30,000
Shim/drive cycles	Normal/upset	1,000

In addition to the above transients, the following transients have been considered in the design and fatigue analysis of the CRD.

Transient	Category	
Scram with inoperative buffer	Normal/upset	24
Operation Basis Earthquake*	Normal/upset	10
Safe Shutdown Earthquake	Faulted	1
Scram with stuck control blade	Faulted	1
Control rod ejection accident	Faulted	1

* The frequency of this transient would indicate emergency category. However, for conservatism this OBE condition was analyzed as upset. Ten peak OBE cycles are postulated.

3.9.1.1.2 CRD Housing and Incore Housing Transients

The number of transients, their cycles, and classification as considered in the design and fatigue analysis of the CRD housing and incore housing are as follows:

Transient	Category	Cycles
Startup and shutdown	Normal/upset	300
Design pressure tests	Normal/upset	43
Loss of feedwater pumps	Normal/upset	10
Relief or safety valve blowdown	Normal/upset	8
Scrams	Normal/upset	180
Operation Basis Earthquake (OBE)*	Normal/upset	10
Safe Shutdown Earthquake (SSE)**	Emergency CRD HSG	1
	Faulted #Incore HSG	1
Stuck rod scram CRD HSG	Normal/upset	1
Scram no buffer only	Normal/upset	24

* The frequency of this transient would indicate emergency category. However, for conservatism this OBE condition was analyzed as upset. Ten peak OBE cycles are postulated.

** SSE is faulted condition: However, in the CRD housing stress analysis report, it was treated as an emergency condition with lower stress limits, thus making the comparison of results to the more conservative allowable.

3.9.1.1.3 Hydraulic Control Unit Transients

The transients used in the design and analysis of the Hydraulic Control Unit and its components are:

Transient	Category	Cycles
Reactor startup/shutdown	Normal/upset	120
Scram test	Normal/upset	140
Startup scrams	Normal/upset	160
Operational scrams	Normal/upset	300
Jog cycles	Normal/upset	30,000
Shim/drive cycles	Normal/upset	1,000
Scram with stick scram discharge valve	Emergency	1
OBE*	Normal/upset	10
SSE	Faulted	1

* The frequency of this event would indicate an emergency category. However, for conservatism this OBE condition was analyzed as normal and upset event. Ten peak OBE cycles are postulated.

3.9.1.1.4 Core Support and Reactor Internals Transients

The cycles listed in Table 3.9-1 were considered in the design and fatigue analysis for the reactor internals. As part of the 120% Extended (Licensed) Power Uprate (LPU @ 3473 MWt), a structural integrity assessment of the key reactor internal components was performed. The thermal hydraulic analysis data, Reactor Internal Pressure Differences, and the acoustic and flow induced loads due to a postulated Recirculation line break (LOCA), including GE14 fuel, were used as input to the EPU evaluation (Ref. 11).

3.9.1.1.5 Main Steam System Transients

The following transients are considered in the stress analysis of the main steam piping:

Transient	Category	Cycles
Hydrotest	Test	40
Leaktest	Test	360
Startup	Normal	120
Turbine trip	Upset	10
Scram and trip isolation valves open	Upset	40
Scram	Upset	140
Shutdown	Normal	111
Composite loss of feedwater, loss of auxiliary power and turbine trip without bypass	Upset	10

Transient	Category	Cycles
Turbine bypass single relief or safety valve	Upset	8
Reactor over pressure delayed scram	Emergency	1
Automatic blow down	Emergency	1
Operating Basis Earthquake (OBE)*	Upset/normal	50
Turbine stop valve closure (TSV)	Upset	660
Relief Valve Lift (RLV)	Upset	5433

* The frequency of this event would indicate an emergency category. However, for conservatism this OBE condition was analyzed as normal and upset event. Fifty peak OBE cycles are postulated.

3.9.1.1.6 Recirculation System Transients

The following transients are considered in the stress analysis of the recirculation piping:

Transient	Category	Cycles
Hydrotest	Test	40
Startup	Normal	120
Turbine Trip	Upset	10
Partial feedwater heater bypass	Upset	70
Turbine generator trip, F.W. on, isolation valves open	Upset	40
Scram	Upset	140
Shutdown	Normal	111
Composite loss of feedwater pumps and auxiliary power, and turbine generator trip without bypass	Upset	10
Turbine bypass single S/RV blowdown	Upset	8
Reactor overpressure with delayed scram	Emergency	1
Automatic depressurization system actuation	Emergency	1
Operating Basis Earthquake (OBE)*	Upset/normal	50
Single Loop Operation	Normal	50

* The frequency of this event would indicate an emergency category. However, for conservatism this OBE condition was analyzed as normal and upset event. Fifty peak OBE cycles are postulated.

3.9.1.1.7 Reactor Assembly Transients

The reactor assembly includes.the reactor pressure vessel support skirt, shroud support, and shroud plate. The cycles listed in Table 3.9-1 were specified in the reactor assembly design and fatigue analysis.

3.9.1.1.8 Main Steamline Isolation Valve Transients

The main steamline isolation valves are designed for the following service conditions and thermal cycles:

Transient	Category	Cycles
Startup and shutdown		
a. Heating cycle @100° F/hr	Normal/upset	300
b. Cooling cycle @100° F/hr	Normal/upset	300
c. ±29° F step change between 70° F and 552° F	Normal/upset	600
d. ±50° F step change between 70° F and 552° F	Normal/upset	200
Loss of feedwater pump/MSIV closure		
a. 552° F to 573° F in 3 seconds (ΔT = 21° F heating)	Normal/Upset	10
b. 573° F to 525° F in 9 minutes (ΔT = 48° F cooling)	Normal/Upset	10
c. 525° F to 573° F in 6 minutes (ΔT = 48° F heating)	Normal/Upset	10
d. 573° F to 485° F in 7 minutes (ΔT = 88° F cooling)	Normal/Upset	10
e. 485° F to 573° F in 8 minutes (ΔT = 88° F heating)	Normal/Upset	10
f. 573° F to 485° F in 7 minutes (ΔT = 88° F cooling)	Normal/Upset	10
Single relief valve blowdown		
a. 552° F to 375° F in 10 minutes (ΔT = 177° F cooling)	Normal/Upset	8
Reactor overpressure with delayed scram		
a. 552° F to 586° F in 2 seconds (ΔT = 34° F heating)	Emergency	1
b. 586° F to 561° F in 30 seconds (ΔT = 25° F cooling)	Emergency	1
Automatic blowdown (ADS)		
a. 552° F to 375° F in 3.3 minutes (ΔT = 177° F cooling)	Emergency	1
b. 375° F to 259° F in 19 minutes (ΔT = 116°F cooling)	Emergency	1
Pipe rupture and blowdown		
a. 552° F to 259° F in 15 seconds (ΔT = 293° F cooling)	Faulted	1

3.9.1.1.9 Safety/Relief Valve Transients

The transients used in the analysis of the safety/relief valves are as follows:

Transient	Category	Cycles
Heating and cool-down - within the temperature limits of 70° F and 552° F at a rate of 100° F/hr	Normal/upset	300
Small temperature changes - of 29° F (either increase or decrease) at any temperature between the limits of 70° F and 552° F.	Normal/upset	600
50° F temperature changes - (either increase or decrease) at any temperature between the limits of 70° F and 552° F.	Normal/upset	200
Loss of feedwater pumps, isolation valve closure	Normal/upset	10
Turbine bypass, single relief or safety valve blowdown (temperature drops from 552° F to 375° F in 10 minutes).	Normal/upset	8
Reactor overpressure with delay scram - (temperature rise from 552° F to 586° F in 2 seconds, and the pressure rises from 1050 psig to 1375 psig, immediately followed by a cooling transient in which the temperature drops from 586° F to 561° F in 30 seconds and the pressure drops to 1125 psig)	Emergency	1
Automatic blowdown - (temperature changes from 552° F to 375° F in 3.3 minutes, immediately followed by a change from 375° F to 259° F in 19 minutes)	Emergency	1

Pipe rupture and blowdown - (temperature drops from 552° F to 259° F in 15 seconds)	Faulted	1
Installed hydrotests - valve inlet nozzle and disc shall be designed to withstand the following:		
a. hydrotests to 1045 psig at 100° F	Testing	120
b. steamline flooding during plant shutdown	Other	120

Eight cycles of an SRV discharge event are conservatively postulated as explained in response to Q&R Item No. 55. The blowdown assumed for an ADS actuation is more rapid than would occur with a single SRV stuck open. Analysis shows that the probability of an inadvertent ADS actuation is 10^{-4} to 10^{-2} per year. Therefore, one inadvertent ADS cycle for the 40-year plant life is conservative. (Q&R MEB (DSER) 57)

Paragraph NB3552 of ASME III code excludes various transients and provides means for combining those which are not excluded. Review and approval of the equipment suppliers certified calculations provides assurance of proper accounting of the specified transients.

3.9.1.1.10 Recirculation Flow Control Valve Transients

The following pressure and temperature transients were considered in the design of the recirculation system flow control valve:

Transient	Category	Cycles
Startup/Shutdown (100°F/hr heatup/cooldown rate between 70°F design temperature)	Normal/upset	350
Small temperature changes (±29°F step	Normal/upset	600
±50°F step changes)	Normal/upset	200
Safety/relief valve blowdowns (single valve) (552°F to 375°F in 10 minutes)	Normal/upset	8
Safety valve transient (110% of design pressure)	Normal/upset	1
Installed hydrotests (at 100°F)		
1300 psig	Testing	130
1670 psig	Testing	3
Automatic blowdown 552°F to 375°F in 3.3 minutes and 375°F to 281°F in 19 minutes	Emergency	1
Improper start of pump in cold loop (100°F to 552°F over a period of 15 seconds)	Emergency	1

3.9.1.1.11 Recirculation Pump Transients

The following pressure and temperature transients were considered in the design of the Recirculation pump.

Transient	Category	Cycles
Startup/Shutdown (100°F/hr heatup/cooldown rate between 70°F design temperature)	Normal/upset	300
Small temperature changes (±29° F step	Normal/upset	600
±50° F step changes)	Normal/upset	200
Safety/relief valve blowdowns (single valve) (552° F to 375° F in 10 minutes)	Normal/upset	8

Safety valve transient (110% of design pressure)	Normal/upset	1
Installed hydrotests (at 100° F)		
1300 psig	Testing	130
1670 psig	Testing	3
Automatic blowdown 552° F to 375° F in 3.3 minutes and 375° F to 281° F in 19 minutes	Emergency	1
Improper start of pump in cold loop (100° F to 552° F over a period of 15 seconds)	Emergency	1

3.9.1.1.12 Recirculation Gate Valve Transients

The following transients are considered in the design of the recirculation gate valves:

Transient	Category	Cycles
Startup/Shutdown (100°F/hr heatup/cooldown rate between 70°F design temperature)	Normal/upset	350
Small temperature changes (±29°F step	Normal/upset	600
±50°F step changes)	Normal/upset	200
Safety/relief valve blowdowns (single valve) (552°F to 375°F in 10 minutes)	Normal/upset	8
Safety valve transient (110% of design pressure)	Normal/upset	1
Installed hydrotests (at 100°F)		
1300 psig	Testing	130
1670 psig	Testing	3
Automatic blowdown 552°F to 375°F in 3.3 minutes and 375°F to 281°F in 19 minutes	Emergency	1
Improper start of pump in cold loop (100°F to 552°F over a period of 15 seconds)	Emergency	1
Pipe rupture and blowdown (552°F to 281°F in 15 minutes)	Fault	1

3.9.1.1.13 Balance-Of-Plant (BOP) System and Component Design Transients

The list of transients used in the design and fatigue analysis of ASME Code Class 1 components and component supports is presented in Tables 3.9-1 and 3.9-1(a). A description of transient load of a dynamic nature, associated with plant and system operating conditions, is presented in more detail in Subsection 3.9.3 and Attachment A3.9.

The definitions of the design transients are consistent with those specified in the design specifications.

The number of cycles considered for fatigue evaluation based on OBE earthquake loads is specified in Subsection 3.7.3.2.1.

The individual system design transients are based on the applicable plant events and associated cycles defined in Tables 3.9-1 and 3.9-1(a), which are based on a 40 year design life.

Portions of the piping subsystem 1MS-38A, (outside containment-main steam drain line), have a more limiting design life, as defined in the Piping Design Specification.

The list of thermal transients identified in Table 3.9-1(a) are based on those established for the RPV nozzles and are tabulated for the major piping systems connected to the RPV. While these transients represent the most severe conditions for piping analysis, the BOP system analysis may incorporate modified transients based on actual system operation. Use of modified transients will be documented in the Certified Stress Report.

3.9.1.2 Computer Programs Used in Analysis

The following sections discuss computer programs used in the analysis of specific components. (Computer programs were not used in the analysis of all components, therefore not all components are listed). Subsections 3.9.1.2.1 through 3.9.1.2.5, 3.9.1.3.3, and 3.9.1.4.3 reference computer programs utilized by GE and vendors for analyzing NSSS components. These NSSS programs can be divided into two categories:

GE Programs

The verification of the following GE programs has been performed in accordance with the requirements of 10 CFR 50, Appendix B. Evidence of the verification of input, output, and methodology is documented in GE Design Record Files.

a. SEISM
b. MASS
c. SNAP
d. ANSYS
e. CREEP-PLAST
f. PISYS
g. ANSI7
h. RVFOR
i. TSFOR
j. PDA
k. SAP4G
l. FTFLGOI
m. POSUM
n. BILRD
o. DYSEA
p. SPECA
q. CRDSSOI
r. ISCOR09
s. LAMB07
t. ECGEN02

Vendors' Programs

The verification of the following two groups of vendor programs is assured by contractual requirements between GE and the vendors. Per the requirements, the quality assurance procedure of these proprietary programs used in the design of N-Stamped equipment is in full compliance with 10 CFR 50, Appendix B.

Byron-Jackson Programs

a. RTRMEC
b. FMAP
c. FLTFLG

d. MULTISPAN
e. 2DFMAP
f. CRISP

CB&I Programs

a. GENOZZ
b. NAPALM
c. 1027
d. 846
e. 781-KALNINS
f. 979-ASFAST
g. 766-TEMAPR
h. 767-PRINCESS
i. 928-TGRV
j. 962-E0962A
k. 984
l. 992-GASP
m. 1037-DUNHAMS
n. 1335
o. 1606 and 1657-HAP
p. 1635
q. 953
r. 1666
s. 1684
t. E1702A
u. 955-MESHPLOT
v. 1028
w. 1038

Subsection 3.9.1.2.6 describes computer programs used by Sargent & Lundy and Structural Integrity Associates in analysis of components.

3.9.1.2.1 Reactor Vessel and Internals

3.9.1.2.1.1 Reactor Vessel

The computer programs used in the preparation of the reactor vessel stress report are identified and their use summarized in the following paragraphs. (The RPV test is applicable only to plants with RPV fabricated by CB&I and CBIN).

3.9.1.2.1.1.1 CG&I Program 7-11- "GENOZZ"

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, Subsection NB 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 acceptable results.

3.9.1.2.1.1.2 CB&I Program 9-48- "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. The program analyzes at specified locations from the point of application of 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 the angular location around the circumference of the nozzle from the 0 reference 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.

3.9.1.2.1.1.3 CB&I Program 1027

This program is a computerized version of the analysis method contained in the "Welding Research Council Bulletin F107, Dec 65.

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 the determination of each S, there are also determined the components of that S (two normal stresses, sx and sy, and one shear stress t). This program provides the same information as the manual calculation and the input data is essentially the geometry of the vessel and attachment.

3.9.1.2.1.1.4 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, Subsection NB.

In cases where the penetration has a counterbore, the thickness is determined so that the counterbore does not penetrate the outside surface of the head.

3.9.1.2.1.1.5 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.

The basic method of analysis was published by Professor Kalnins in the Journal of Applied Mechanics, Volume 31, September, 1964, pages 467 through 476.

3.9.1.2.1.1.6 CB&I Program 979 - "ASFAST"

ASFAST Program (Program 979) performs the stress analysis of axisymmetric, bolted closure flanges between head and cylindrical shell.

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 axisymmetric loading:

a. preload condition,

b. internal pressure, and

c. thermal load.

3.9.1.2.1.1.7 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. The program is written in FORTRAN IV language.

3.9.1.2.1.1.8 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, Subsection NB, of the ASME Pressure Vessel Code.

3.9.1.2.1.11.9 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.

The program utilizes an electrical network analogy to obtain the temperature distribution of any given system as a function of time. The finite difference representation of the three-dimensional equations of heat transfer are repeatedly solved for small time increments and continually summed. Linear mathematics is used to solve the mesh network for every time interval. 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, its geometry and 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.

3.9.1.2.1.1.10 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.

3.9.1.2.1.1.11 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, Subsection NB, of the ASME Code.

3.9.1.2.1.1.12 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 and is written in FORTRAN IV. For a detailed account, see the following reference document:

Wilson, E. L.; "A Digital Computer Program for the Finite Element Analysis of Solids with Non-Linear Material Properties" Aerojet General Corporation, Sacramento, California. Technical Memorandum No. 23, July 1965.

As mentioned above, the program determines the stresses and displacements of plane or axisymmetric structures using the finite element method. The structures may have arbitrary geometry and have linear or non-linear 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 of "nodal-points" or "nodes." The actual loads on the structure are simulated by statically equivalent loads acting at the appropriate nodes. The basic input to the program consists of the geometry of the stress-model and the boundary conditions. The program then gives the stress components at the center of each element and the displacements at the nodes, consistent with the prescribed boundary conditions.

3.9.1.2.1.1.13 CB&I Program 1037 - "DUNHAM'S"

DUNHAM'S program is a finite ring element stress analysis program. It will determine the stresses and displacements of axisymmetric structures of arbitrary geometry subjected to either axisymmetric loads or non-axisymmetric loads represented by a Fourier series.

This program is similar to the GASP program (CB&I 992). The major differences are that DUNHAM'S can handle non-axisymmetric loads (which requires that each node have three degrees of freedom) and the material properties for DUNHAM'S must be constant. As in GASP, the loadings may be thermal, mechanical, and accelerational.

3.9.1.2.1.1.14 CB&I Program 1335

To obtain stresses in the shroud support, the baffle plate must be made a continuous circular plate. The program makes this modification and allows the baffle plate to be included in CB&I program 781 as two isotropic parts and an orthotropic portion at the middle (where the diffuser holes are located).

3.9.1.2.1.1.15 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.

3.9.1.2.1.1.16 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.

3.9.1.2.1.1.17 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.

3.9.1.2.1.1.18 CB&I Program 1666

This program is written primarily to calculate the temperature differences at selected Critical Sections of the Nuclear Reactor Vessel components at different time points of thermal transients during its life of operation and list them all in a tabular form. Since there is no involved calculation applicable particularly to nuclear components, this program can be used with any other kind of model that sees thermal transients over a period of time. This program helps ascertain the time points in thermal transients when thermal stresses may be critical.

3.9.1.2.1.1.19 CB&I Program 1684

This program is written to better expedite the fatigue analysis of nuclear reactor components as required by the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. Specifically, this program is an expansion of an earlier program, 984. The features of this program allow the user to easily perform the complete secondary stress and fatigue evaluation including partial fatigue usage calculation of a component in one run. An additional option allows the user to completely document the input stress values in a format suitable for a stress analysis report. The program is written to allow for a minimum amount of data handling by the user once the initial deck is established.

3.9.1.2.1.1.20 CB&I Program "E1702A"

This program evaluates the stress-intensity factor KI due to pressure, temperature, and mechanical load stresses for a number of different stress conditions (times) and at a number of different locations (elements). It then calculates the maximum RTNDT which the actual material can have based on a 1/4T flaw size and compares it with the ordered RTNDT. If the ordered RTNDT is larger than the maximum RTNDT, the maximum allowable flaw size is calculated. The rules of Appendix G are used except that WRC 175 can be used to calculate KI due to pressure in a nozzle to shell junction.

For a more thorough description of the fracture problem, see WRC Bulletin No. 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials."

3.9.1.2.1.1.21 CB&I Program 955 "MESHPLOT"

This program plots input data used for finite element analysis. The program plots the finite element mesh 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.

3.9.1.2.1.1.22 CB&I Program 1028

The program calculates the necessary form factors for the nodes of the model transfering heat by radiation when given the shape and dimensions of the head-to-skirt knuckle junction model. Program is limited to junction which has a toroidal knuckle part.

3.9.1.2.1.1.23 CB&I Program 1038

The program calculates the loads required in order to satisfy the compatibility between the shroud baffle plate and the jet pump adaptors for a GE BWR vessel.

3.9.1.2.1.2 Reactor Internals

3.9.1.2.1.2.1 Fuel Support Loads Program/SEISM

SEISM computes the vertical fuel support loads using the component element methods in dynamics. The methodology is based on the publication, "The Component Element Methods in Dynamics", by S. Leveg and J. D. P. Wilkinson, McGraw Hill Co., New York, 1976.

3.9.1.2.1.2.2 Other Programs

The following computer programs are used in the analysis of the core support structures and other safety-related reactor internals: MASS, SNAP (MULTISHELL), ANSYS, and CREEP-PLAST. Details of these programs are provided in Section 4.1.

3.9.1.2.2 Piping

3.9.1.2.2.1 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 Problem," NEDO-24210, August, 1979.

3.9.1.2.2.2 Component Analysis/ANSI 7

3.9.1.2.2.2.1 Application

The ANSI 7 Computer Program determines stress and accumulative usage factors for thermal, weight, seismic relief valve lift and turbine stop valve closure (as applicable) conditions of loadings derived from the Structural System Analysis in accordance with NB-3600 of ASME Boiler and Pressure Vessel Code Section III.

3.9.1.2.2.2.2 Program Organization

For Class 1 stress analysis the program generates and prints hoop, bending, thermal discontinuity, linear temperature gradient and nonlinear temperature gradient components of stress for each equation of subarticle NB-3600 of Section III. Load combination results from possible load sets for Class 1 stress equations. The total stress (sum of component stresses) and the stress ratio (total stress divided by appropriate stress intensity limit) is printed for each Class 1 equation. The total stress (sum of component stresses) and the stress ratio (total stress divided by the appropriate stress intensity limit) is printed for each one of the equations 9, 10, 12 and 13 of NB-3600. The alternating stresses and usage factor are calculated per NB-3653.6. Verification of this program is discussed in Subsection 3.9.1.2.

3.9.1.2.2.3 Relief Valve Discharge Pipe Forces Computer Program/RVFOR

The relief valve discharge pipe connects the relief valve to the suppression pool. Under normal circumstances, the discharge end of the pipe is under water and the remainder of the pipe is filled with air. The water may be drawn up into the pipe if the air inside is at less than atmospheric pressure. 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.

3.9.1.2.2.4 Turbine Stop Valve Closure/TSFOR

The TSFOR program computes the time history forcing function in the main steam piping due to turbine stop valve closure. The program utilizes the method of characteristics to compute fluid momentum and pressure loads at each change in pipe section or direction.

3.9.1.2.2.5 Piping Dynamic Analysis Program/PDA

The pipe whip analysis was performed using the PDA computer program (Reference 9). 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 timedependent 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.

3.9.1.2.3 Pumps and Motors

3.9.1.2.3.1 Recirculation Pump Program (ANSYS)

The ANSYS code using finite element methods is used in the analysis of the recirculation pump casing for various thermal and mechanical loads during plant operating and postulated conditions.

In general, the finite element techniques are used to solve temperature distribution in heat transfer transient problems, and to perform stress analysis for various thermal and mechanical loadings by using the same finite element model representing the pump body. The output of these programs is in the form of temperature profiles, deflections, and stresses at the nodal points of the finite element idealization of the pump structure.

3.9.1.2.3.2 ECCS Pumps and Motors (Byron Jackson Programs)

RTRMEC is a computer program which calculates and displays results of mechanical analysis of motor rotor assembly when acted upon by external forces at any point along the shaft (rotating parts only). The shaft deflection analysis including magnetic and centrifugal forces was analyzed. The calculation for the seismic condition assumes that the motor is operating and that the seismic, magnetic, and centrifugal forces all act simultaneously and in phase on the rotor-shaft assembly. Note that the distributed rotor assembly weight is lumped at the various stations, with the shaft weight at a station being the sum of one-half the weight of the incremental shaft length just before the station, plus one-half the weight of the adjacent incremental shaft length just after the station. Bending and shear effects are accounted for in the calculations.

FMAP is a computer program which solves for the natural frequencies and associated mode shapes of a two-dimensional frame. The frame is defined as a system of uniform, weightless members whose ends or joints are rigidly attached. All weights are lumped at the joints. Each joint has three degrees of freedom: two translations in the plane of the frame and a rotation about the axis normal to the plane The frame is in the X-Y plane, and all motion of the frame is in this plane.

FLTFLG computer program determines stresses in bolted flanged connections where the flanges are flat faced and bolted together directly or separated by a metal spacer such that there is metal to metal contact beyond the bolt circle. Calculation procedure follows rules set forth in Appendix II, Part B, ASME Boiler and Pressure Vessel Code Section VIII, Division 1, 1971, Winter 1973 Addenda.

MULTISPAN is a computer program which performs the bending analysis of variable cross-section continuous beams up to ten spans. The analysis yields reactions, internal forces, displacements, and maximum shear and bending stresses.

2DFMAP is a computer program which solves for the natural frequencies and the associated mode shapes of a rigidly jointed, two-dimensional lumped-mass frame. The solution is based on small-deflection theory assuming linear stiffnesses for the frame. Stiffness matrix alternations can be used to add complex structural elements which cannot be represented by members. Gaussian elimination is available to reduce the size of the stiffness matrix if relatively small weights are associated with any degree of freedom. The frequencies and mode shapes are computed using the Householder Strum method and inverse iteration.

CRISP computer program determines the fundamental and harmonic modes of lateral vibration of rotating elements of arbitrary flexural rigidity. The computational method is based on a transfer matrix representation of the rotor shaft which includes the effect of multiple supports with dissimilar elasticity and damping in the bearings and with dissimilar elasticity and mass of the bearing supports. In addition to calculating the natural frequencies, the program provides lateral deflections for the determination of rotor stresses, running clearances, and severity of vibration at the different resonant frequencies. Vibration amplitudes of the bearing supports are also provided for determining support resonant frequencies and for obtaining an optimum design through modifications of the bearing and their supports.

3.9.1.2.3.3 ECCS Pumps and Motors (GE Programs)

3.9.1.2.3.3.1 Structural Analysis Program/SAP4G

SAP4G is used to analyze the structural and functional integrity of the ECCS pump/motor systems. This is a general structural analysis program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve the displacements and stresses of each element of the structure. The structure can be composed of unlimited numbers of three-dimensional truss, beam, plate, shell, solid, plate strainplane stress and spring elements that are axisymmetric. The program can treat thermal and various forms of mechanical loading. The dynamic analysis includes mode superposition, time history, and response spectrum analysis. Seismic loading and time-dependent pressure can be treated. The program is versatile and efficient in analyzing large and complex structural systems. The output contains displacements of each nodal point as well as stresses at the surface of each element.

3.9.1.2.3.3.2 Effects of Flange Joint Connections/FTFLGOI

The flange joints connecting the pump bowl castings are analyzed using FTFLGOI. This program uses the local forces and moments determined by SAP4G to perform flat flange calculations in accordance with the rules set forth in Appendix II and Section III of the ASME Boiler and Pressure Vessel Code.

3.9.1.2.3.3.3 Beam Element Data Processing/POSUM

POSUM is a computer code designed to process SAP generated beam element data for pump or heat exchanger models. The purpose is to determine the load combination that would produce the maximum stress in a selected beam element. It is intended to be used on RHR heat exchangers with four nozzles or ECCS pumps with two nozzles.

3.9.1.2.4 RHR Heat Exchangers

3.9.1.2.4.1 Structural Analysis Program/SAP4G

SAP4G is used to evaluate the structural and functional integrity of the RHR heat exchangers. A description of this program is provided in Subsection 3.9.1.2.3.3.1.

3.9.1.2.4.2 Calculation of Shell Attachment Parameters and Coefficients/BILRD

BILRD is used to calculate the shell attachment parameters and coefficients used in the stress analysis of the support to shell junction. The method, per Welding Research Council Bulletin No. 107, is implemented in BILRD to calculate local membrane stress due to the support reaction loads on the heat exchanger shell.

3.9.1.2.4.3 Beam Element Data Processing/POSUM

POSUM is used to process SAP generated beam element data. The description of this program is provided in Subsection 3.9.1.2.3.3.3.

3.9.1.2.5 Dynamic Loads Analysis

3.9.1.2.5.1 Dynamic Analysis Program/DYSEA

DYSEA simulates a beam model in the annulus pressurization dynamic analysis. A detailed description of DYSEA is provided in Section 4.1. DYSEA employs a preprocessor program named GZAPL. GZAPL converts pressure, time histories into time varying loads and forcing functions for DYSEA. The overall resultant forces and moments time histories at specified points of resolution can also be obtained from GZAPL.

3.9.1.2.5.2 Acceleration Response Spectrum Program/SPECA

SPECA generates acceleration response spectrum for an arbitrary input time history of piece-wire linear accelerations, i.e., to compute maximum acceleration responses for a series of single-degree-of-freedom systems subjected to the same input. It can accept acceleration time histories from a random file. It also has the capability of generating the broadened/enveloped spectra when the spectral points are generated equally spaced on a logarithmic scale axis of period/frequency. This program is also used in seismic and SRV transient analyses.

3.9.1.2.6 Computer Programs Used in the Analysis of Balance-of-Plant Systems and Components

The computer programs used in the analysis of mechanical systems and components within the scope of the A/E are identified in Subsections 3.9.1.2.6.1 through 3.9.1.2.6.25.

The computer program used by Structural Integrity Associates is identified in Subsection 3.9.1.2.6.26.

For each computer program, there is a brief discussion of the theoretical basis, assumptions, and limitations, extent of application and validation. All programs have been verified as to correctness of theory used and the validity of results with the stated assumptions and limitations as required by applicable quality assurance requirements.

3.9.1.2.6.1 DYNAX

TITLE: Dynamic Analysis of Axisymmetric Structures (DYNAX)

For program information, see Appendix C of the CPS-USAR.

3.9.1.2.6.2 ENV

TITLE: Response Spectra Envelope (ENV)

PROGRAM NO.: 09.5.135

AUTHOR: M. Nopola, EMD

PROGRAM SCOPE: The dynamic analysis of piping systems and equipment whose anchors are located at different elevations and/or buildings requires a combination of spectra for these elevations and/or buildings, resulting in one envelope spectrum. The user may specify one of three ways of generating the envelope: 1) the largest acceleration values, 2) SRSS of the acceleration values, or 3) the absolute sum of all the acceleration values.

VALIDATION: Program was validated by manual calculations for computer output in EMD-021349. The same sample problem was executed for this validation on RUNID SMNENV dated July 8, 1983.

3.9.1.2.6.3 HYTRAN

TITLE: Hydraulic Transient Analysis (HYTRAN)

PROGRAM NO.: 09.5.121

AUTHORS: C. H. Li and V. K. Verma

PROGRAM SCOPE: HYTRAN calculates pressures, velocities, and force transients in a liquid-filled piping network with up to 60 legs of 40 nodes or 200 legs of 15 nodes each. Transients may be initiated by valve closure, pump trip or startup or by pressure changes at a piping terminal. The pump characteristics may be described using two methods, polynomial input or trigonometric input. Sets of data are provided in HYTRAN for pump-specific speeds of 1800, 7600, and 13500. These data may be chosen and then modified to match available data, or the entire set of data may be input.

Output of force-time history can be plotted and/or saved on a data file for use as input to PIPSYS.

VALIDATION: The program was validated by comparison with the following problems:

a. Hydraulic Transients; V. L. Streeter and E.EB. Wylie, 1967 (Problems 3.1, 3.4, and 3.6)

b. Waterhammer Analysis, John Parmakian, 1963 (Problem on pg. 83)

c. Transient Analysis of Offshore Loading Systems, V. L. Streeter & E. B. Wylie, Transactions of the ASME Journal of Engineering Industry, February 1975.

3.9.1.2.6.4 LSS

TITLE: Local Stresses in Spherical and Cylindrical Shells Due to External Loadings on Nozzles (LSS)

PROGRAM NO.: 09.5.117

AUTHOR: A. J. Weiss

PROGRAM SCOPE: This program uses the Bijlaard method of stress analysis to calculate stresses due to external loading on nozzles described in the Welding Research Council Bulletin 107. All the empirical curves in Bulletin 107 were put into equation form, using the curve-fitting Program POLYFIT (09.5.130-1.0).

VALIDATION: LSS was validated by manual calculation. All equations generated by POLYFIT were also checked by comparing the values generated by the equations with the original input data.

3.9.1.2.6.5 LUG

TITLE: Analysis of Lug Supports on Piping System (LUG)

PROGRAM NO.: 09.5.167

AUTHOR: C. A. Podczerwinski, T. G. White, and R. P. Jasinski

PROGRAM SCOPE: This is a system of seven programs which perform stress analysis for welded attachment locations on all classes of piping systems, including fatigue analysis for Class 1 welded attachments. Geometry data for the 12 lug types is supplied interactively and the restraint load data is obtained from the PIPSYS save file. The analysis is performed usingthe methods described in Welding Research Council Bulletin 198, (1974) and the Sample Analysis of a Class 1 Piping System published by 1 ASME, (1972).

VALIDATION: LUG was validated by manual calculations.

3.9.1.2.6.6 NOHEAT

TITLE: Nonlinear Heat Transfer Analysis of Axisymmetric Solids (NOHEAT)

PROGRAM NO.: 09.5.075

AUTHORS: I. Farhoomand and E. L. Wilson, University of California, Berkeley, California

PROGRAM SCOPE: The program uses the finite element method to calculate the temperature distribution in an axisymmetric solid which results from nonlinear heat transfer. The nonlinear effects of conduction, radiation and convection can be included. A temperature history for each node point is presented. Internal generation has been provided for several of the most frequently used meshes. In addition, stresses resulting from linear thermal expansion are calculated for certain appropriate sections. Options have been added which calculate linearized thermal gradients and which plot the finite-element mesh.

VALIDATION: Two problems have been selected to validate this program. The first is taken from "ASME/Pressure Vessel and Piping/1972 Computer Program Verification" and is problem AER-I, "An Axisymmetrical Transient Thermal Analysis."

The second problem is a straight length of pipe subject to an internal temperature change of 432°F in 0.5 second. This problem was solved using both NOHEAT and TSHOK (09.5.033) and the results were compared.

3.9.1.2.6.7 PENAN

TITLE: Penetration Assembly Stress Analysis Program (PENAN)

PROGRAM NO.: 09.5.134-1.3

AUTHORS: S. R. Taylor and A. P. Dimopoulos

PROGRAM SCOPE: The program handles structural, thermal, and fatigue analyses of specialized penetration assemblies. The analyses are performed to the requirements set forth in ASME B&PV Code, Section III, Division 1.

VALIDATION: The major analytic capabilities of the program were validated using hand calculations and/or existing validated computer programs. The following computer programs were used for the validation:

a. DYNAX (09.7.083-7.1).

b. NOHEAT (09.5.075-3.1).

3.9.1.2.6.8 PESSAL

TITLE: Penetration Design Program (PESSAL)PROGRAM NO.: 09.5.086-4.1

AUTHORS: S. R. Taylor, R. M. Tjernlund, and A. P. Dimopoulos

PROGRAM SCOPE: The computer program PESSAL calculates penetration sleeve thicknesses, faulted condition loads, and cooling coil requirements. In addition, it maintains a data file containing the information required to prepare two design reports: the Penetration Sleeve Design Report and the Penetration Assembly Design Report.

VALIDATION: PESSAL was validated by hand calculations.

3.9.1.2.6.9 PIPSYS

TITLE: Integrated Piping Analysis System (PIPSYS)

For program information, see Appendix C of the CPS-FSAR.

3.9.1.2.6.10 PWRRA

TITLE: Pipe Whip Restraint Reaction Analysis (PWRRA)

PROGRAM NO.: 09.5.125-1.0

AUTHORS: M. A. Salmon and V. K. Verma

PROGRAM SCOPE: This program computes the maximum response by the energy balance method of a simplified model of a pipe-restraint support structure system to a loading consisting of an initial impulse plus a suddenly applied steady load. Displacements of the pipe, restraint, and structure as well as the restraint reaction time history are given as output.

VALIDATION: The program was validated by comparing results for a series of 14 problems with those given by the finite-difference program PWRRA (09.5.125-2.1).

3.9.1.2.6.11 PWRRA

TITLE: Pipe Whip Restraint Reaction Analysis (PWRRA)

PROGRAM NO.: 09.5.125-2.1

AUTHORS: V. K. Verma and M. A. Salmon

PROGRAM SCOPE: This program computes the maximum response to a time-dependent forcing function of a simplified model of the pipe and restraint system for the purpose of designing pipe whip restraints.

VALIDATION: The program has been validated on the basis of available analytical results published in technical literature and GAAA Report Number VII 1289 dated June 2, 1976.

3.9.1.2.6.12 PWUR

TITLE: Pipe Whip for Unrestrained Pipes (PWUR) PROGRAM NO.: 09.5.137

AUTHOR: J. J. Lula

PROGRAM SCOPE: This program performs an analysis of a rupture of an unrestrained pipe on the surrounding area. Three types of analysis are available: circumferential break impact, longitudinal break impact, and jet impingement.

VALIDATION: PWUR was validated by hand calculations.

3.9.1.2.6.13 RELVAD

TITLE: Relief Valve Design Program (RELVAD)

PROGRAM NO.: 09.5.136

AUTHOR: J. J. Lula

PROGRAM SCOPE: The computer program RELVAD is used to assist the engineer while designing safety/relief valve assemblies. The program calculates fluid forces at valve discharge exit and vent stack inlet and exit, moments and stresses in the discharge elbow, discharge flange, valve inlet weld and branch connection to the run. In addition, the program considers four methods of designing the branch connections: 1) weldolet, 2) sweepolet, 3) constant wall thickness tee, 4) constant inner diameter tee.

VALIDATION: Validation by comparison to the example problem in American National Standard Code for Pressure Piping Winter 1975 Addenda to Power Piping ANSI B31.1g-1976.

3.9.1.2.6.14 RESGEO

TITLE: Geometry of Pipe Whip Restraints (RESGEO)

PROGRAM NO.: 09.5.106

AUTHOR: J. A. Diaz

PROGRAM SCOPE: RESGEO calculates the geometry of pipe whip restraints with respect to a global coordinate system, giving the true length of legs and the angles between them, and the point of attachment on the vessel or containment wall.

VALIDATION: RESGEO was validated by manual calculation of representative geometries. The program output results were assumed as given and the input was calculated.

3.9.1.2.6.15 SIPDA

TITLE: Simplified Piping Dynamic Stress Analysis (SIPDA)

PROGRAM NO.: 09.5.047

AUTHORS: E. B. Branch, M. Nopola

PROGRAM SCOPE: This program develops a set of conservative design curves that are used to:

a. Select spans whose first period is removed from the predominant building period.

b. Ascertain that stresses do not exceed allowables.

c. Assure that the deflections do not result in contact between piping and surroundings.

d. Provide restraint design loads.

VALIDATION: Sample problems were validated by manual calculations.

3.9.1.2.6.16 SLSAP4

TITLE: Sargent & Lundy Structural Analysis Program (SLSAP4)

For program information, see Appendix C of the CPS-FSAR.

3.9.1.2.6.17 SRVA

TITLE: Safety/Relief Valve Blowdown Analysis (SRVA)

PROGRAM NO.: 09.5.138

AUTHOR: M. A. Salmon

PROGRAM SCOPE: SRVA is a finite difference program for the analysis of transient flow in a relief valve line discharging to the suppression pool. Transient forces and the pressures at the water column and the valve outlet are calculated for relief valve lines with up to 20 straight segments. Frictional effects as well as losses at elbows and the outlet are included.

Output force-time data is compatible with PIPSYS, and force-time history is plotted by a plot subroutine.

VALIDATION: This program was validated by two methods:

a. Comparing the pressures with those published in the General Electric Company report, No. NEDO-10859 dated April 1973.

b. Comparison with analytical solutions of instantaneous valve opening with sonic and subsonic flow at the entrance.

3.9.1.2.6.18 TSHOK

TITLE: Thermal Shock Stress in Cylinders and Restrained Plates (TSHOK)

PROGRAM NO.: 09.5.033-2.0

AUTHORS: E. B. Branch and M. Nopola

PROGRAM SCOPE: This program calculates the transient thermal gradient stresses induced by transient thermal behavior in the contained fluid. It also computes the time-dependent mean wall temperatures and two time-dependent quantities, T_1 and T_2, which are required for Class A piping stress analysis, as specified in ASME Boiler and Pressure Vessel Code, Section III, 1975, Subsection NB-3600.

VALIDATION: TSHOK was validated by hand calculation and by the program NOHEAT.

3.9.1.2.6.19 PIPERUP

TITLE: Pipe Force And Whip Analysis (PIPERUP)

PROGRAM NO.: 09.5.201

AUTHOR: Nuclear Service Corporation, Division of Quadrex

PROGRAM SCOPE: PIPERUP is used for analysis of piping system force and whip for consideration of postulated ruptures. It is a finite element computer code that performs nonlinear elastic-plastic analysis of three dimensional piping systems subjected to concentrated static or dynamic time history forcing functions. These forces may result from fluid jet thrust at the location of a postulated rupture of high-energy piping. Straight and curved beam (elbow) elements are used for mathematical representation of piping, and axial and rotational springs are used to represent restraints. The stiffness characteristics of piping and restraints can reflect elastic/linear strain hardening material properties and gaps between restraints can be modeled.

The program was developed by Nuclear Service Corporation division of Quadrex and it is implemented on Control Data computers (Cyber-175 and Cyber-176). Since the program is not used a great deal at Sargent & Lundy, when it is used, a service bureau having CDC computers and licensed by Quadrex to offer PIPERUP services where PIPERUP is used UCC (University Computing Company) and CDC Cybernet.

VALIDATION: The program was validated by comparison of sample problem results with those presented in the "PIPERUP Computer Program Verification Manual" (Quadrex, QUAD-7-81-048).

3.9.1.2.6.20 ANSYS4

TITLE: Engineering Analysis System (ANSYS4)

PROGRAM NO.: 09.5.185

AUTHOR: Swanson Analysis Systems, Inc.

PROGRAM SCOPE: ANSYS4 is the 1982 version of a proprietary software product of Swanson Analysis Systems, Inc. of Houston, Pennsylvania which was licensed to Sargent & Lundy in accordance with Purchase Order No. 26040 dated July 15, 1982. It is a finite element program originally developed in 1970, which has application to many engineering analysis problems including structural engineering (static and dynamic analysis; elastic, plastic, creep, and swelling considerations; small and large deflection modeling), heat transfer (steady-state and transient analysis; conduction, convection, and radiation), and piping analysis. ANSYS4 has one-, two-, and three-dimensional capability for axisymmetric or planar coordinate systems. Coupled thermohydraulic, thermoelectric, and wave motion capability are available in ANSYS4.

Data to be input to ANSYS4 are quite extensive and problem dependent with matrix format in files generally applied to the preprocessing stage. See Chapter 3 of the user's manual for a complete description of input preparation.

Output data are handled by any post processors including one dealing with a general database and a variety of others from shell and plate element post processors, response spectrum generators, and piping system evaluators. Plotting capability is also available.

VALIDATION: The program was validated by comparison of results obtained from 125 sample problems with those presented in Swanson's "ANSYS Engineering Analysis System Verification Manual," 3-1-82.

3.9.1.2.6.21 AXTRAN

TITLE: Axial Temperature Transients In Welds (AXTRAN)

PROGRAM NO.: 09.5.181

AAUTHOR: M. Veg

PROGRAM SCOPE: Axtran solves the heat transfer problem of a semi-infinite solid subjected to surface temperature variations with time. The TA-TB temperatures are computed after initial and boundary conditions are applied to the heat conduction equation. The nominal code stresses are arrived at by averaging the temperature distribution for each point over a predetermined region.

VALIDATION: The program was validated by comparison of sample problem results with those produced by validated program NO HEAT (Prog. no. 09.5.075).

3.9.1.2.6.22 NONLIN2

TITLE: Nonlinear Dynamic Analysis of Two Dimensional Structures (NONLIN2)

For Program Information, See Appendix C of the CPS-USAR

3.9.1.2.6.23 RELAP4

TITLE: A Computer Program For Transient Thermal-Hydraulic Analysis Of Nuclear Reactors And Related Systems (RELAP4)

PROGRAM NO.: 09.8.026

AUTHOR: Idaho National Engineering Laboratory (INEL)

PROGRAM SCOPE: RELAP4 is a computer program that was developed to describe the thermal-hydraulic behavior of light-water reactor systems subjected to postulated transients such as those resulting from loss-of-coolant, pump failure, or nuclear power excursions. RELAP4 can also be used to calculate the behavior of a part of a system provided the appropriate thermal-hydraulic boundary condition inputs are made to the program. RELAP4 is comprehensive and predicts the interrelated effects of coolant thermal-hydraulics, system heat transfer, and core neutronics. Because the program was developed to solve a large variety of problems, the user must specify the applicable program options and the system to be analyzed.

The RELAP4 program controls are used to specify the problem dimensions and constants, time step size, trip controls for reactor system transient behavior, and output. Controls are also provided for restarting a problem and producing a plotting tape. There are four basic options in the RELAP4 code that can be selected by input. These are Standard RELAP4, RELAP4-EM, RELAP4-FLOOD, and RELAP4-CONTAINMENT.

The major parts of the RELAP4 program are the fluid equations (Section 3.3), heat transfer (Section 3.4) and reactor kinetics (Section 3.5). These are outlined as follows:

The fluid dynamics portion of RELAP4 solves the fluid mass, energy, and flow equations for the system being modeled. In order to provide a reasonable degree of versatility, a choice of the following five basic forms of the flow equation is provided:

a. Compressible single-stream flow with momentum flux

b. Compressible two-steam flow with one-dimensional momemtum mixing

c. Incompressible single-stream flow without momentum flux

d. Compressible integral momentum

e. Incompressible mechanical energy balance.

The compressible two-stream flow equation has four sub-forms to represent different stream flow patterns.

The fluid system to be analyzed by RELAP4 must be specified by the user. It must be modeled by fluid volumes and by fluid junctions (flow paths) between the volumes. User specified fluid volumes (control volumes) are used to represent the fluid in the system associated with a heat sink or source, such as fuel rods or a heat exchanger. The fluid volumes are connected by junctions which are used to transfer fluid into and out of fluid volumes. Junctions are of three types:

a. Normal (connects fluid volumes)

b. Leak (system fluid loss point)

c. Fill (system fluid gain point)

A junction must be located within the elevations specified for the fluid volumes that are connected to the junction because the fluid path is physically continuous. A normal junction connects two adjacent fluid volumes.

A heat conductor model is used to transfer heat to or from the fluid in a volume. The geometry and conditions of the heat conductor are specified by the user. This model may be used to describe the thermal behavior and effects of fuel rods, pipes, and plates. The program contains correlations for calculating the critical heat flux (CHF), pre-CHF heat transfer, and post-CHF heat transfer. Several options are also available for describing heat exhangers.

Program options are available for describing the power internally generated in system components such as fuel rods or electric heaters. These options include user-supplied normalized power versus time curves and program solution of the space-independent reactor kinetics equations with or without radioactive decay heat.

VALIDATION: The program was validated using a set of sample problems supplied with the code. The output was compared to results generated by the IBM/370 version maintained at the Argonne Code Center.

3.9.1.2.6.24 RFC

TITLE: Response Fatigue Cycles (RFC)

PROGRAM NO.: 09.5.194

AUTHOR: Z. N. Ibrahim

PROGRAM SCOPE: RFC determines the occurrence response equivalent number of fatigue cycles at the maximum component response level which is generally used to evaluate the component sustained load capacity. These equivalent fatigue cycles in conjunction with the maximum response amplitude would produce the same fatigue damaging effect of the full response history variable amplitude stress reversals. The program can also be used to determine the base excitation displacement consistent with the base excitation acceleration. Options are available to obtain the statistical analysis of input excitation history, associated consistent displacement and dynamic response relative displacement. The program is capable of performing fatigue damage calculations for combined occurrences by both the probabilistic approach and the common cycle elimination technique. Plots of the input time histories, consistent displacement, dynamic response and their pertinent CDF's are optional.

VALIDATION: The program was validated by comparison with closed form solutions, hand calculations and with the validated program (PIPSYS (09.5.065).

3.9.1.2.6.25 TRANS

TITLE: TRANS-Translating Of Forces And Moments To A Common Point.

PROGRAM NO.: 09.5.017

AUTHORS: S. Holtsman, M. Nopola

PROGRAM SCOPE: This program translates the forces and moments from one or more points to one common point and gives the resultant forces and moments at all points.

VALIDATION: The program was validated by manual calculations.

3.9.1.2.6.26 VESLFAT

VESLFAT is a program that has been used in ASME Section III, Subsection NB-3200 evaluation of BOP piping

3.9.1.3 Experimental Stress Analysis

Subsections 3.9.1.3.1 through 3.9.1.3.3 list the only NSSS components upon which experimental stress analysis was used and provide a discussion of the analysis. The applicability of experimental stress analysis for BOP components is addressed in Subsection 3.9.1.3.4.

3.9.1.3.1 Experimental Stress Analysis of Piping Components

The following components have been tested to verify their design adequacy:

a. Piping snubbers.

b. Pipe whip restraints.

Descriptions of the support and whip restraint tests are contained in Subsection 3.9.3.4 and Section 3.6, respectively.

3.9.1.3.2 Orificed Fuel Support, Vertical and Horizontal Load Tests

The BWR 6 Orificed Fuel Support (OFS) under the provisions of the ASME Code, Section III, Subsection NG is classified as a core support structure and, therefore, must comply with NA-3352.1. In order to meet this requirement, an analysis was performed using the finite element method. However, the complexity of the OFS design as well as the nonlinear behavior of the OFS during analysis preempted the use of finite element analysis. Accordingly, a series of horizontal and vertical load tests were performed in order to conform to the requirements of the code and the design specification. The results of these tests indicate that the hydrodynamic and seismic loading of the OFS are below the load limit allowables determined in accordance with NG-3228.4 and F-1380. Allowable limits are .44 L_u for upset and .80 P_t for faulted, where L_u and P_t are the loads used in testing.

Differences which may exist between the actual parts and the tested parts including dimensional thickness, yield strength and casting quality are accounted for. The comparison of calculated and allowable loads is shown in Table 3.9-2(b).

3.9.1.3.3 Control Rod Drive

Experimental data was used in refining the CRDSSOI code. The output of CRDSSOI was used in the dynamic analysis of both code 1 and noncode parts. Pressures used in the analysis were also determined during actual testing of prototype control rod drives.

As an example, the CRD internal transient pressures were recorded during the various drive functional tests. These pressure traces represent the pressure characteristics during scram, jogs, and shim/drive cycles. The pressure traces were incorporated into the CRD stress analysis along with other loads (thermal, mechanical).

3.9.1.3.4 Experimental Stress Analysis for BOP Systems and Components

Experimental stress analysis as described in Appendix II of the ASME B&PV Code will not be used for any Seismic Category I systems and components for which S&L has design and analysis responsibility. For systems and components for which the design and/or analysis and Certified Stress Report are the responsibility of the component manufacturer, the use and the justification of experimental analysis methods will be documented in the Certified Stress Report

3.9.1.4 Considerations for the Evaluation of Faulted Conditions

For NSSS components, all Seismic Category I equipment is evaluated for the faulted loading conditions and in all cases the calculated stresses are within the specified allowables. The following paragraphs in Subsections 3.9.1.4.1 through 3.9.1.4.12 show examples of the treatment of faulted conditions for the major components on a component by component basis. Additional discussion of faulted analysis can be found in Subsections 3.9.3 and 3.9.5 and Table 3.9-2.

Subsection 3.9.2.2 and Section 3.7 discuss the seismic and hydrodynamic events treatment of dynamic loads resulting from the postulated SSE. Subsection 3.9.2.5 discusses the dynamic analysis of loads on NSSS equipment resulting from blowdown. Deformations under faulted conditions have been evaluated in critical areas and no cases are identified where design limits, such as clearance limits, are violated.

3.9.1.4.1 Control Rod Drives System Components

3.9.1.4.1.1 Control Rod Drives

The major CRD components that have been analyzed for the faulted conditions are: ring flange, main flange, and indicator tube.

The maximum stress for these components for a faulted event at various operating conditions is provided in Table 3.9-2(u).

The faulted design stress limits for Class 1 components are specified in the ASME Code, Section III, Appendix F (F-1323 lb). The primary stress limits are the lesser of the following:

(a) Membrane: $2.4S_m$ or $0.70S_u$

(b) Membrane + Bending: $1.5\ (2.4S_m) = 3.6S_m$

or $1.5\ (0.70S_u) = 1.05S_u$

The membrane stress limit is less than the ultimate stress (S_u). However, the maximum membrane plus bending stress limit could be 5% greater than S_u. The calculated stresses are based on the elastic analysis; this would usually overestimate the actual stress of the part. Unlike a part subjected to tensile loads, the component in bending would not result in area reduction. Furthermore, the elastically calculated outer fiber bending stress may exceed theoretically the ultimate stress without the collapse of the part, because physically the stresses will be redistributed due to the plastic flow in the outer fibers and the maximum fiber stress will not reach the ultimate limit. The bulk of the material will still remain in the elastic range.

The applicable code for the CRD is the 1971 edition, Summer 1973 addenda. Appendix F was introduced in Winter 1972 addenda and the elastic limit on local membrane plus bending stress was not changed in the Summer 1973 addenda. (Q&R MEB (DSER) 64)

3.9.1.4.1.2 Hydraulic Control Unit

The hydraulic control unit (HCU) was analyzed for the seismic and hydrodynamic load condition. Subsection 3.9.2.2.1.6.4 describes the method of this analysis.

3.9.1.4.1.3 CRD Housing

The CRD housing is analyzed for the faulted condition. The SSE and hydrodynamic loads are considered.

Table 3.9-2v shows the load combinations, analytical methods and allowable and calculated stress values for the highly stressed areas of the control rod drive housing.

RCI used four computer programs in the CRD piping analysis. The four programs were TPIPE Version 5.1, EASE 2 Version 13.4, E2A17 Version 13.4B, and EWELD Version 2.0. The description of each computer program including the extent of its application and the method of verification is shown in Attachment C3.9. (Q&R 210.04)

3.9.1.4.2 Standard Reactor Internal Components

3.9.1.4.2.1 Control Rod Guide Tube

The maximum calculated stress on the CR guide tube occurs in the base during the faulted condition. The faulted limit is 2.4 S_m per ASME Code Section III Table I-1.2 where S_m = 16,000 psi at 575° F.

The analysis and results for various plant conditions are summarized in Table 3.9-2(y).

3.9.1.4.2.2 Incore Housing

The maximum calculated stress on the incore housing occurs at the outer surface of the vessel penetration during the faulted condition. The faulted limit is the lesser of 2.4 Sm or 0.7 Su at the design temperature per ASME Code Section III F323.1(b). Table 3.9-2(z) shows that the calculated stresses are within the allowables.

3.9.1.4.2.3 Jet Pump

An elastic analysis for the jet pump at faulted conditions shows that the maximum stress is due to impulse loading of the diffuser during a pipe rupture and blowdown. The maximum allowable for this condition per ASME Code Section III, Subsection NG is 3.6 S_m. Table 3.9-2(w) shows that the calculated stresses are within the allowables.

3.9.1.4.2.4 LPCI Coupling

The maximum stress on the LPCI coupling during the faulted condition is bounded by the allowable of 3.6 x 0.7 x S_m, where a weld quality factor of 0.7 is included as required by the ASME Code Section III, Table NG-3352-1. The analysis and results are summarized in Table 3.9-2 (x) for various plant conditions.

3.9.1.4.2.5 Orificed Fuel Support

See Subsection 3.9.1.3.2.

3.9.1.4.3 Reactor Pressure Vessel Assembly

The reactor pressure vessel assembly includes the reactor pressure vessel, shroud support, and support skirt.

For faulted conditions the reactor pressure vessel assembly was evaluated using elastic analysis. For the reactor, ultimate strength allowable values were not used since the emergency allowable stress limits of ASME Section III, Subsection NB were used for the faulted condition. For the shroud support and support skirt, an elastic analysis was performed, except for the support legs, for primary membrane stress, and for a compressive loading case where buckling was evaluated. For this analysis the Creep-Plast computer program was used, which is described in Subsection 4.1.4.1.10. Verification of this program has been performed as described in Subsection 3.9.1.2. The analysis results are summarized in Table 3.9-2(a).

3.9.1.4.4 Core Support Structure

The core support structures are evaluated for the faulted condition on the basis .of the seismic and other dynamic events described in Section 3.7 and Subsection 3.9.5, respectively. The calculated stresses and allowables are summarized in Table 3.9-2(b).

3.9.1.4.5 Main Steam Isolation, Recirculation Gate and Safety/Relief Valves

Tables 3.9-2(g), (h), and (j) provide a summary of the analysis of the safety/relief, main steam isolation and recirculation gate valves, respectively.

Standard design rules, as defined in ASME Code, Section III, Subsection NB, are utilized in the analysis of pressure boundary components of Seismic Category I valves. Conventional elastic stress analysis is used to evaluate components not defined in the ASME Code. The code allowable stresses are applied to determine acceptability of structure under applicable loading conditions including the faulted condition.

3.9.1.4.6 Recirculation System Flow Control Valve

The recirculation system flow control valve is analyzed for faulted conditions using the elastic analysis criteria from the ASME Code, Section III, Subsection NB-3500.

The analysis is summarized in Table 3.9-2(f).

3.9.1.4.7 Main Steam and Recirculation Piping

For main steam and recirculation system piping, elastic analysis methods are used for evaluating faulted loading conditions. The equivalent allowable stresses using elastic techniques are obtained from ASME Code allowable stresses obtained from ASME Code Section III, Appendix F, "Rules for Evaluation of Faulted Conditions," and these are greater than the elastic limits. Additional information on the main steam and recirculation piping and pipe-mounted equipment is in Tables 3.9-2(d) and (e).

3.9.1.4.8 Nuclear Steam Supply System Pumps, Heat Exchanger, and Turbine

The recirculation, ECCS, RCIC, and SLC pumps, the RHR heat exchangers and the RCIC turbine have been analyzed for the faulted loading conditions identified in Subsection 3.9.1.1. In all cases, stresses were within the elastic limits. The analytical methods, stress limits, and allowable stresses are discussed in Subsections 3.9.2.2 and 3.9.3.1.

3.9.1.4.9 Control Rod Drive Housing Supports

The calculated stresses and stress limits for faulted conditions for the control rod drive housing supports are shown in Table 3.9-2(t).

3.9.1.4.10 Fuel Assembly (Including Channels)

GE BWR fuel assembly design bases, analytical methods and evaluation results, including those applicable to the faulted conditions, are contained in References 4 and 5. The resulting combined acceleration under various loading conditions is less than the corresponding evaluation basis as shown in Table 3.9-2(b).

3.9.1.4.11 Reactor Refueling and Servicing Equipment

Items of refueling and servicing equipment which are important to safety are classified as essential components per the requirements of 10 CFR 50, Appendix A. Equipment whose failure would degrade an essential component is defined in Section 9.1 and is classified as Seismic Category I. These components are subjected to an elastic dynamic finite element analysis to generate loadings. This analysis utilizes appropriate seismic floor response spectra and combines loads at frequencies up to 33 hertz for seismic and up to 60 Hz for hydrodynamic loads in three directions. Imposed stresses are generated and combined for normal, upset, and faulted conditions. Stresses are compared, depending on the specific safety class of the equipment, to Industrial Codes, ASME, ANSI, industrial standards, or AISC allowables. The calculated stresses and allowables for the fuel storage rack, refueling platform, fuel preparation machine, and fuel transfer tube are shown in Table 3.9-2(s).

3.9.1.4.12 Considerations for the Evaluation of the Faulted Condition for BOP Systems and Components

Static and dynamic loads associated with faulted conditions are evaluated using dynamic load factors, structural response spectra, or time history analysis and assume elastic behavior of the component. A component is assumed to behave elastically if yielding across a section does not occur. The limits of the elastic range are defined in Paragraph F-1323 of Appendix F of the ASME B&PV Code. Local yielding due to stress concentrations is assumed not to affect the validity of the assumption of elastic behavior.

Evaluations under faulted conditions, based on Service Level D limits, generally apply to non-essential components or as specified in Attachment A3.9.

In those cases where component stresses exceed yield as a direct result of pipe rupture, simplified methods of analysis based on conservation of momentum and energy are used. When simplified methods are used, the component is designed to absorb energy without exceeding maximum allowable strain limits based on material properties and stress states.

Where simplified methods of analysis do not adequately characterize the system or component response, an elastic-inelastic time history analysis is performed. Maximum allowable strain limits are based on material types and stress states.

3.9.2 Dynamic Testing and Analysis

3.9.2.1 Piping Vibration, Thermal Expansion and Dynamic Effects

Asymmetric LOCA loads are being considered for design of these systems, and in Subsection 6.2.1.2 of the USAR a discussion on these loads is presented.

Item 1

Pedestal plans and details are shown on Figure 3.8-26, Sheets 1 and 2, The pedestal is made of ASTM A588 steel. A detailed description is given in Subsection 3.8.3.1.4. Details of the RPV are given in Subsections 5.3.3.1 and 5.3.3.2.

Item 2

A specific plant analysis has been performed. Details of this analysis are presented in the following items.

Item 3

Breaks of concern are shown in the following figures:

a. Steam line B3.6-6, B3.6-7, B3.6-8, and B3.6-9

b. Feedwater B3.6-1 and B3.6-2

c. Recirculation B3.6-18

The break analyses are discussed in 6.2.1.2. Evaluation of the effects of these breaks are presented in Items 6 and 7 below.

Item 4

For BOP and NSSS structures and components no inelastic action results from the application of these loads.

Item 5

The starting point for all the annulus pressurization analyses performed as part of the design of the biological shield wall and piping within the biological shield annulus is the thermal-hydraulic analysis that determines the pressures vs. time within the annulus as the results of postulated pipe ruptures. These analyses are discussed in detail in Subsections 6.2.1.2.1.2, 6.2.1.2.2.2. and 6.2.1.2.3.2. The USAR discusses the analysis of the reactor recirculation line break and the feedwater line break.

The general arrangement of the biological shield annulus and enclosed major piping is shown in Drawings M01-1111-4, M27-1314, and Figures 6.2-57 through 6.2-59. The analytical models are discussed in Subsection 6.2.1.2.3.2 and shown graphically in Figures 6.2-21, 6.2-22, 6.2-85, 6.2-86, 6.2-148 and 6.2-149. The mathematical model data is presented in Table 6.2-14, Tables 6.2-21 through 6.2-23, and Tables 6.2-67 through 6.2-68. The mass/energy release rates are determined by the method described in GE document APED-4827 and tabulated in Tables 6.2-41, 6.2-42 and Table 6.2-69. These analyses were performed using the RELAP4 and WARLOC computer codes as discussed in Subsection 6.2.1.2.3.2. The results of the analyses are shown as pressure-time histories for the various spatial modes in Figures 6.2-23 through 6.2-54, 6.2-87 through 6.2-93 and 6.2-150 through 6.2-178.

A sensitivity analysis has been performed to demonstrate the adequacy of these analyses the details of which are discussed in USAR Subsection 6.2.1.2.3.2.

An axisymmetric finite element model of the reactor pressure vessel, shield wall and pedestal has been used to calculate responses of these structures to the pressure loads discussed in Subsection 6.2.1.2.3.2.

For BOP the computer program DYNAX, described in Appendix C, has been used to calculate dynamic responses through direct numerical integration. Separate analyses are performed for each postulated break. The pressure time histories are applied at the nodes as equivalent nodal force time histories. The responses are calculated using the unbalanced pipe break pressure loads within the shield wall annulus. Acceleration response spector developed through the use of structural response time histories and the computer program RSG, described in Appendix C, or response time histories are used directly in analyzing piping systems excited by the annulus pressurization loads through the supports and anchors which are attached to the pedestal, shield wall or the RPV.

For NSSS the pressure time histories are converted to equivalent beam nodal force histories using the GEAPL computer program. These force histories are then used as input to a beam model for response determination using the DYSEA program. The beam modeling is consistent with the procedures described in Section 3.7. The results of this beam analysis are used for analysis and code evaluation of the RPV internals. In addition, the pressure time histories are also represented in Fourier series for use in the ASHSD computer program that analyzes the shell model of the RPV and internals plus the pedestal and shield wall. This model is illustrated in Figure 3.9-12. The output consists of loads for use in code evaluation of the RPV stresses in accordance with the load combination given in Table 3.9-2. Acceleration response spectra are also obtained from the output acceleration time histories. The response spectra of acceleration time histories were obtained for the analysis of main steam and recirculation piping.

For BOP the computer program used in the piping analysis due to the AP loads is PIPSYS (Appendix C) for both response spectra, and multiple acceleration time history method of analysis. Modeling of the piping system is the same as that explained in Subsection 3.7.3.3.1.1.

The piping model and method of analysis for NSSS are per Subsection 3.7.3.3.1.2. The dynamic analysis for piping was performed using the PISYS program (refer to Subsection 3.9.1.2).

The evaluation of the results for governing load combinations was performed using the ANS17 computer program (refer to Subsection 3.9.1.2).

Allowable stresses for structures and components are covered in Section 3.8 and 3.9.

Item 6

The structural integrity of the safety-related components listed above is maintained when they are subjected to the combined loads resulting from LOCA and SSE (refer to Sections 3.8 and 3.9).

Item 7

Functional capability requirements of all essential piping are met as noted in note 1 of Table A3.9-6. (Q&R MEB (DSER) 67B)

3.9.2.1.1 Piping Vibration and Dynamic Effects (NSSS)

The remainder of Subsection 3.9.2.1 is historical:

The test program is divided into five general phases: preoperational thermal expansion, preoperational piping vibrations, preoperational piping dynamic effects, startup thermal expansion, and startup piping vibration and dynamic effects.

3.9.2.1.1.1 Piping Vibration

3.9.2.1.1.1.1 Preoperational Vibration Testing of Recirculation Piping

The purpose of the preoperational vibration test phase is to verify that operating vibrations in the recirculation piping are within acceptable limits. This test uses visual observation and supplemental remote measurements. If, during steady state operation, visual observation indicates that vibration is significant, measurements will be made with hand held instruments. Visual observations, manual and remote measurements will be made during the following steady-state conditions:

a. recirculation pumps minimum flow;

b. recirculation pumps at 50% of rated flow;

c. recirculation pumps at 75% of rated flow; and

d. recirculation pumps at 100% of rated flow.

3.9.2.1.1.1.2 Preoperation Vibration Testing of Small Attached Piping

During visual observation of steady state recirculation pump minimum flow, 100% of rated flow and an intermediate value, special attention will be given to small attached piping and instrument connections to ensure that they are not in resonance with the recirculation pump motors or flow induced vibrations. If the operating vibration acceptance criteria are not met, corrective action such as modification of supports will be taken.

3.9.2.1.1.1.3 Startup Vibration Testing of Main Steam and Recirculation Piping

The purpose of this phase of the program is to verify that vibration of the main steam and recirculation piping is within acceptable limits. Because of potentially high temperature and radiation levels, visual observation will be supplemented with remote instrumentation. Visual observations, manual, and remote measurements will be made during the following steady state conditions:

a. main steam flow at 25% of rated;

b. main steam flow at 50% of rated;

c. main steam flow at 75% of rated;

d. main steam flow at 100% of rated; and

e. main steam piping at main steam flow at 120% of original rated power level.

3.9.2.1.1.1.4 Operating Transient Loads on Recirculation Piping

The purpose of the operating transient test phase is to verify that pipe stresses are within ASME Section III, Subsection NB Code Limits. The amplitude of displacements and number of cycles per transient of the recirculation piping will be measured and the displacements compared with acceptance criteria. The deflections are correlated with stresses to verify that the pipe stresses remain within Code limits. Visual observation with manual vibration and deflection measurements shall be made during the following transients:

a. recirculation pump starts;

b. recirculation pump trip at 100% of rated flow;

3.9.2.1.1.1.5 Operating Transient Loads on Main Steam Piping

The purpose of the operating transient test phase is to verify that pipe stresses are within ASME Section III, Subsection NB Code Limits. The amplitude of displacements and number of cycles per transient of the main steam piping will be measured and the displacements compared with acceptance criteria. The deflections are correlated with stresses to verify that the pipe stresses remain within Code limits. Visual observation with manual vibration and deflection measurements (supplemented with remote instrumentation) shall be made during the following transients:

a. Turbine stop valve closure at 100% of rated flow;

b. Manual discharge of each SRV at 1000 psig and at planned transient tests that result in SRV discharge.

3.9.2.1.1.2 Dynamic Effects Testing of Main Steam and Recirculation Piping

To verify that snubbers are adequately performing their intended function during plant operation, a program for a dynamic testing is planned as a part of normal startup operation testing. The main purpose of this program is to ensure the following:

a. The vibration levels from the various dynamic loadings during transient and steady-state conditions are below the maximum acceptable limits.

b. Long-term fatigue failure will not occur due to underestimating the dynamic effects caused by cyclic loading during planned transient operations.

This dynamic testing is to account for the acoustic wave due to the safety/relief valve lifts, (RV1), safety/relief valve load resulting from air clearing (RV2), and turbine stop valve closure load (TSVC). The maximum stresses developed in the piping by the RV1, RV2, and TSVC transient analysis is used as a basis for establishing a criterion which will assure proper functioning of the snubbers. If field measurements exceed criteria limits, this may indicate that snubbers are not operating properly. If field measurements are within criteria limits, it will be assumed that snubbers are functioning properly. Sample production snubbers of each size (i.e., 10 kips, 20 kips, 50 kips, etc.) shall also be qualified and tested for design and faulted condition loadings prior to shipment to field. Snubbers shall be tested to allow free piping movements at low velocity. During plant startup, the snubbers shall be checked for improper settings and checked for any evidence of hydraulic fluid leakage.

The criteria for vibration displacements shall be based on assumed linear relationship between displacements, snubber loads and magnitude of applied loads for any function and response of system. Thus the magnitude of limits of displacements, snubber loads, and nozzle loads are all proportional. Maximum displacements (Level 1 limits) are established to prevent the maximum stress in the piping systems from exceeding the normal and upset primary stress limit and/or the maximum snubber load from exceeding the maximum load to which the snubber has been tested.

Based on the above criteria, Level 1 displacement limits are established for all instrumented points in the piping system.

These limits shall be compared with the field measured piping displacements. Method of acceptance shall be as explained in Subsection 3.9.2.1.3.

3.9.2.1.2 Piping Vibration and Dynamic Effects (Non-NSSS)

Tests shall be conducted for piping vibration, thermal expansion, and dynamic effects for designated piping in the following essential systems:

a. MSIV Leakage Control

b. Reactor Core Isolation Cooling

c. High Pressure Core Spray

d. Low Pressure Core Spray

e. Standby Liquid Control

f. Residual Heat Removal

3.9.2.1.2.1 Piping Vibration

3.9.2.1.2.1.1 Preoperational Vibration Testing (Other than NSSS Scope)

During preoperational testing of each of the systems as listed in Table 3.9-13, the system will be visually observed for vibration. If visually perceivable vibration occurs, measurements will be made with portable instruments, and the measurements will be compared with the acceptance criteria.

Lines are included as described in Subsection 3.9.2.1. Due to the number of Main Steam and Reactor Level sensing lines and their inaccessibility during operation, a sampling program may be used. This would be reflected in the test procedures, which will be submitted prior to testing. (Q&R MEB (DSER) 88)

3.9.2.1.2.1.2 Startup Vibration Testing (Non-NSSS Scope)

During startup, Main Steam, Feedwater, RCIC steam supply line, and essential instrument lines for Startup Steady State Vibration Test listed in Table 3.9-1 will be visually monitored for vibration or instrumented where necessary. The inaccessible areas of the feedwater piping system, inside containment, will be instrumented with remote vibration transducers and monitored during startup and initial plant operation at the operational conditions described in

Section 3.9.2.1.1.1.3. The systems will be tested at operating conditions. The measurements will be compared with the acceptance criteria.

3.9.2.1.2.1.3 Startup Operating Transient Loads (Non-NSSS Scope)

The feedwater and main steam piping will be instrumented during the operational transients listed in Table 3.9-13. The other systems shown in the table will be visually checked.during the transients listed.

3.9.2.1.3 Test Evaluation and Acceptance Criteria (NSSS)

In order to ensure test safety, deflection criteria. based upon piping displacement, will be established based on final stress analysis and provided in the startup test specifications. This criteria will be designated Levels 1 and 2 as described in the following paragraphs.

3.9.2.1.3.1 Level 1 Criteria

Level 1 establishes the maximum limits for the level of pipe motion which, if exceeded, makes a test hold or termination mandatory. If the Level 1 limit is exceeded, the plant will be placed in a satisfactory hold condition, and the responsible piping design engineer will be advised. Following resolution, applicable tests must be repeated to verify that the requirements of the Level 1 limits are satisfied.

3.9.2.1.3.2 Level 2 Criteria

Level 2 specifies that the level of pipe motion which, if exceeded, require that the responsible piping design engineer be advised. If the Level 2 limit is not satisfied, plant operating and startup testing plans would not necessarily be altered.

Investigations of the measurements, criteria, and calculations used to generate the pipe motion limits would be initiated. An acceptable resolution must be reached by all appropriate and involved parties, including the responsible piping design engineer. Depending upon the nature of such resolution, the applicable tests may or may not have to be repeated.

3.9.2.1.3.3 Acceptance Criteria (NSSS)

For steady-state vibration, the piping peak stress (zero to peak) due to vibration only (neglecting pressure) will not exceed 10,000 psi for Level 1 criteria and 5,000 psi for Level 2 criteria. These limits are below the piping material fatigue endurance limits as defined in Design Fatigue Curves in Appendix I of ASME Code for 10^6 cycles.

For operating transient vibration, the piping bending stress (zero to peak) due to operating transient only will not exceed $1.2S_m$ or pipe support loads will not exceed the Service Level D ratings for Level 1 criteria. The $1.2S_m$ limit ensures that the total primary stress including pressure and dead weight will not exceed bases $1.8S_m$, the new Code Service Level B limit. Level 2 criteria are on pipe stresses and support loads not to exceed design basis predictions. Design basis criteria require that operating transient stresses and loads not exceed any of the Service Level B limits including primary stress limits, fatigue usage factor limits, and allowable loads on snubbers.

If all Level 2 criteria are satisfied for both steady-state vibration and operating transient vibration there will be no fatigue damage to the piping system due to steady-state vibration and all operating transient vibrations are less than the calculated values in the stress report. If any Level 2 limits are not satisfied, detailed engineering evaluation is needed to develop corrective action or to show that the measured results are acceptable. Any resolution must be properly documented and approved as described in Subsection 3.9.2.1.3.2.

3.9.2.1.4 Test Evaluation and Acceptance Criteria (Non-NSSS)

The following acceptance criteria have been established for steady-state vibration testing and operational transient vibration testing. These criteria are based upon piping displacement amplitude measurements.

3.9.2.1.4.1 Vibration Testing

If, in the judgement of the test engineers, there is no significant vibration of the piping system during applicable mode of operation, the system is acceptable. If there is vibration judged to be significant, the worst displacement amplitude will be measured and compared with an allowable displacement. This allowable displacement will be based on the peak stress values given below.

The rationale for the level of peak stress is that the allowable stress amplitude, Sa used for steady state piping vibration is equal to 80% of the alternating stress intensity at 10^6 cycles for carbon steels and to 60% of the alternating stress intensity at 10^6 cycles for stainless steels. The values of alternating stress intensity are taken from Figures I-9.1 and I-9.2 of Appendix I, ASME Code Section III.

In addition, a "factor of safety" of 1.3 is applied to the allowable stress amplitudes used to determine the vibratory stresses.

Therefore, considering the factor of safety and the allowable stress amplitudes above, the maximum piping stress will be as follows:

$S_a = 0.8 \times 12{,}500/1.3 = 7{,}690$ psi for carbon steels with UTS $\leq$ 80 ksi

$S_a = 0.6 \times 26{,}000/1.3 = 12{,}000$ psi for stainless steels.

If the vibration displacement is not within allowable limits, corrective action in accordance with Subsection 3.9.2.1.5 will be taken.

3.9.2.1.4.2 Operational Transient Vibration Testing

For piping systems that are expected to experience significant operational transients (main steam and feedwater), the measured displacement responses will be compared with the calculated responses from the piping stress report.

For the other systems listed in Table 3.9-13, the piping will be visually inspected. If the test engineer judges the piping response to be unacceptable, the source of the transient will be eliminated, the piping and/or restraints will be modified, or it will be proven that the stresses are acceptable by detailed measurement or analysis.

3.9.2.1.5 Corrective Actions for Vibration and Dynamic Effects Testing

During the course of the tests, the measurements shall be regularly checked to determine compliance with acceptance criteria. If trends indicate that acceptance criteria may be violated, the measurements shall be monitored at more frequent intervals. The test will be held or terminated as soon as acceptance criteria are violated. If measurements indicate failure to meet NSSS Level 1 criteria, the test will be stopped and the following corrective actions will be taken prior to continuing the test:

a. Installation Inspection. A walkdown of the piping and suspension will be made to identify any obstruction or improperly operating suspension components. Snubbers shall be at about the midpoint of the total travel range at operating temperature. Hangers shall be in their operating range between the hot and cold setting. If vibration exceeds criteria, the source of the excitation must be identified to determine if it is related to equipment failure. Action will be taken to correct any discrepancies before repeating the test.

b. Instrumentation Inspection. The instrumentation installation and calibration will be checked and any discrepancies corrected. Additional instrumentation will be added, if necessary.

c. Repeat Test: If actions (a) and (b) identify discrepancies that could account for failure to meet Level 1 criteria, the test will be repeated.

d. Resolution of Findings: If the Level 1 criteria is violated on the repeat test or no relevat discrepancies are identified in (a) and (b), the organization responsible for the stress report shall review the test results and criteria to determine if the test can be safely continued.

If the test measurements indicate failure to meet NSSS Level 2 or BOP criteria, the following corrective actions will be taken after completion of the test:

a. Installation Inspection: A walkdown of the piping and suspension shall be made to identify any obstruction or improperly operating suspension components. Snubbers shall be at about the midpoint of the total range at operating temperature. Hangers shall be in their operating range between the hot and cold settings. If vibration exceeds limits, the source of the vibration must be identified. Action, such as suspension adjustment, will be taken to correct any discrepancies.

b. Instrumentation Inspection: The instrumentation installation and calibration will be checked and any discrepancies corrected.

c. Repeat Test: If (a) and (b) above identify a malfunction or discrepancy that could account for failure to comply with criteria and appropriate corrective action has been taken, the test may be repeated.

d. Documentation of Discrepancies: If the test is not repeated, the discrepancies found under actions (a) and (b) above shall be documented in the test evaluation report and correlated with the test condition. The test will not be considered complete until the test results are reconciled with the acceptance criteria.

3.9.2.1.6 Measurement Locations

3.9.2.1.6.1 Measurement Locations for NSSS Piping

Remote vibration transducer measurement locations will be selected to monitor the vibration levels of the NSSS piping during startup and initial plant operation at the operational conditions described in Section 3.9.2.1.1.1.3. During preoperational testing prior to fuel load, visual inspection of the piping will be made, and any visible vibration measured with a handheld instrument.

For each of the selected remote measurement locations, Level 1 and 2 deflection limits will be prescribed in the startup test specification. Level 2 limits will be based on the results of the stress report adjusted for operating mode and instrument accuracy; Level 1 limits will be based on maximum allowable Code stress limits.

3.9.2.1.6.2 Measurement Locations for Non-NSSS Piping

Measuring instruments and visual observation points will be located generally as described below. Actual instrument locations will be included in the detailed test instructions.

Visual observations and/or vibration measurements will be taken at positions close to the forcing function (i.e., a pump outlet). For those systems which are instrumented during operational transients, dynamic displacement measurements will be taken at the points of the greatest expected pipe motion as described by the pipe design engineers. The component of motion to be measured will be that which is of special interest or is expected to be of critical importance.

Thermal displacement measurements will be taken at positions where displacement will be greatest or where displacement will be judged to be critical.

Temperature measuring devices will be placed in conjunction with other measurement devices where temperature is expected to affect the performance of other measurement devices. These will be used to provide any temperature related corrections that may be necessary to the instruments and/or data.

3.9.2.1.7 Thermal Expansion Testing

A thermal expansion preoperational and startup testing program is performed. Potentiometer sensors, manual measurements, and visual observations are used as specified in Table 3.9-13 to verify that normal thermal movement occurs in the piping systems. The main purpose of this program is to ensure the following conditions:

a. During system heatup and cooldown, the piping system is free to expand and move without unplanned obstruction or restraint in the x, y, and z directions.

b. The piping system is working in a manner consistent with the assumptions of the stress analysis.

c. There is adequate agreement between calculated values of displacements and measured value of displacement.

d. There is consistency and repeatability in thermal displacements during heatup and cooldown of the systems.

Limits of thermal expansion displacements are established prior to start of piping testing to which the actual measured displacements can be compared to determine acceptability of the actual motion. If the measured displacement does not vary from the acceptance limits values by more than the specified tolerance, the piping system is responding in a manner consistent with predictions and is therefore acceptable. Two levels of limits of displacements are established to check the systems as explained in Subsection 3.9.2.1.3.

During this testing, snubbers will be visually observed to verify that they move between the hot and cold positions and are not at the end of their stroke in either hot or cold position. Prior to preoperational testing of snubbers, a preservice examination which incorporates the recommendations of Reference 6, "Preservice Inspection and Testing of Snubbers," is performed. During preoperational testing, snubber thermal movements for systems whose operating temperature exceeds 250°F are verified per the recommendations of Reference 6.

Corrective action will be taken as described in Subsection 3.9.2.1.5.

3.9.2.2 Seismic Qualification of Safety-Related Mechanical Equipment

3.9.2.2.1 Seismic and Hydrodynamic Qualification of Safety-Related NSSS Mechanical Equipment

This subsection describes the criteria for dynamic load qualification of safety-related mechanical 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. These modules are generally discussed in this paragraph rather than providing discussion of the separate electrical parts in Subsections 3.10 and 3.11. Dynamic qualification testing is also discussed in Subsection 3.9.3.2. Electrical supporting equipment such as control consoles, cabinets, and panels which are part of the NSSS are discussed in Subsection 3.10.

3.9.2.2.1.1 Tests and Analysis Criteria and Methods

The ability of equipment to perform its safety-related function during and after the application of dynamic loads is demonstrated by tests and/or analysis. Selection of testing, analysis or a combination of the two is determined by the type, size, shape, and complexity of the equipment being considered. When practical, equipment operability is demonstrated by test. Otherwise, it is demonstrated by mathematical analysis.

Equipment which is large, simple, and/or consumes large amounts of power is usually qualified by analysis or test to show that the loads, stresses and deflections are less than the allowable maximum. Analysis and/or testing is also used to show there are no natural frequencies below 33 Hz for seismic load and 60 Hz for hydrodynamic loads. If a lower natural frequency is determined, dynamic tests and/or analyses are performed to verify operability and structural integrity for the required dynamic input conditions.

When the equipment is qualified by dynamic test, the response spectrum or time history of the attachment point is used in determining input motion.

Natural frequency may be determined by running a continuous sweep frequency search using a sinusoidal steady state input of low magnitude. Dynamic load conditions are simulated by testing using random vibration input or single frequency input (within equipment capability) over the frequency range of interest. Whichever method is used, the input amplitude during testing envelopes the actual input amplitude expected during dynamic load conditions.

The equipment being dynamically tested is mounted on a fixture which simulates the intended service mounting and causes no dynamic coupling to the equipment.

Equipment having an extended structure, such as a valve operator, is analyzed by applying static equivalent dynamic loads at the center of gravity of the extended structure. In cases where the equipment structural complexity makes mathematical analysis impractical, a test is used to determine spring constant and operational capability at maximum equivalent dynamic load conditions. Pipe-mounted equipment is analyzed in the piping system dynamic analysis.

Methods of Seismic Analysis are discussed in Subsection 3.7.3.5.

3.9.2.2.1.2 Random Vibration Input

When random vibration input is used, the actual input motion envelopes the appropriate floor input motion at the individual modes. However, single frequency input, such as sine waves, can be used provided one of the following conditions are met:

a. The characteristics of the required input motion is dominated by one frequency.

b. The anticipated response of the equipment is 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 will envelope the corresponding response spectra of the individual modes.

3.9.2.2.1.3 Application of Input Motion

When dynamic tests are performed, the input motion is applied to one vertical and one horizontal axis simultaneously. However, if the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction, and vice versa, then the input motion is applied to one direction at a time. In the case of single frequency input, the time phasing of the inputs in the vertical and horizontal directions are such that a purely rectilinear resultant input is avoided.

3.9.2.2.1.4 Fixture Design

The fixture design will simulate the actual service mounting and cause no dynamic coupling to the equipment.

3.9.2.2.1.5 Prototype Testing

Equipment testing is conducted on prototypes of the equipment installed in this plant.

3.9.2.2.1.6 Seismic and Hydrodynamic Qualification of Specific NSSS Mechanical Components

The following sections discuss the testing or analytical qualification of NSSS equipment. Dynamic qualification is also described in Subsections 3.9.1.4, 3.9.3.1, and 3.9.3.2.

3.9.2.2.1.6.1 Jet Pumps

A static analysis of the jet pumps was performed and the stresses resulting from the analysis are below the design allowables.

3.9.2.2.1.6.2 CRD and CRD Housing

The dynamic qualification of the CRD housing (with enclosed CRD) was done analytically, and the stress results of their analysis established the structural integrity of these components. Preliminary dynamic tests have been conducted to verify the operability of the Control Rod Drive during seismic and hydrodynamic event. A simulated test, imposing a static bow in the fuel channels, was performed with the CRD functioning satisfactorily.

3.9.2.2.1.6.3 Core Support (Fuel Support and CR Guide Tube)

A detailed analysis imposing dynamic effects due to seismic and hydrodynamic events has shown that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

3.9.2.2.1.6.4 Hydraulic Control Unit (HCU)

The HCU was analyzed for the faulted condition including the effects of seismic and hydrodynamic loads. The design adequacy is determined by test and analysis. With the back side of the HCU's mounted on beams, the dynamic loads are 0 to 2g horizontal and 0 to 23g vertical at the frequency range of 2.5 to 100 Hz. At these frequencies, the maximum HCU capability by test for the dynamic loads is 7 to 12g horizontal and 10 to 23g vertical.

3.9.2.2.1.6.5 Fuel Assembly (Including Channels)

GE BWR fuel assembly design bases, analytical methods and evaluation results, including seismic and hydrodynamic considerations, are contained in References 4 and 5.

3.9.2.2.1.6.6 Recirculation Pump and Motor Assembly

The recirculation pump, including its appurtenances and supports, individually and as an assembly, is designed to withstand the following dynamic loads:

a. The flooded pump motor 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 on pump case and the top of the motor frame.

b. Primary stresses due to horizontal and vertical dynamic forces were considered to act simultaneously to be added directly. Horizontal and vertical dynamic forces were applied at mass centers and equilibrium reactions were determined for motor and pump brackets.

c. Load, shear, and moment diagrams were constructed to scale, using live loads, dead loads, and calculated snubber reactions. Combined bending, tension and shear stresses were determined for each major motor flange bolting, and pump case.

d. The maximum combined tensile stress in the cover bolting was calculated including tensile stress from design pressure.

e. The brackets, on the pump case, were designed to withstand loads resulting from the building dynamic response.

f. Analyses have been completed which demonstrate that the natural frequency of the assembled pump and motor structure under seismic loading is greater than 33 hertz.

3.9.2.2.1.6.7 ECCS Pump and Motor Assembly

A prototype ECCS pump motor assembly has been seismically qualified via a combination of static analysis and dynamic testing. The complete motor assembly has been seismically qualified via dynamic testing, in accordance with IEEE 344-1975. A three-dimensional finite element model of the ECCS pump and motor assembly and its supports is developed and dynamically analyzed using the response spectrum method to verify that the assembly can withstand dynamic loadings. Stresses at critical locations are evaluated and compared with the allowable limits. The results have demonstrated the design adequacy of this assembly.

The qualification of the pump motor while operating under SSE conditions was provided in the form of a static earthquake-acceleration analysis since the natural frequency is above 33 hertz. Under this criterion, the units were considered to be supported as designed and maximum specified vertical and horizontal accelerations being constantly applied simultaneously in the worst case combination and the results of the analysis indicate the pump is capable of sustaining the above loadings without overstressing the pump components.

3.9.2.2.1.6.8 RCIC Pump Assembly

The RCIC pump is a barrel type design on a large cross-section pedestal.

The RCIC pump assembly has been analytically qualified by static analysis for seismic and hydrodynamic 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 90% of the allowable.

3.9.2.2.1.6.9 RCIC Turbine Assembly

The RCIC turbine has been dynamically qualified via a combination of static analysis and dynamic testing. The turbine assembly consists of rigid masses, wherein static analysis has been utilized, interconnected with control levers and electronic control systems, necessitating final qualification via dynamic testing. Static loading analysis has been employed to verify the structural integrity of the turbine assembly, and the adequacy of bolting under operating and dynamic loading conditions. The complete turbine assembly has been qualified via dynamic testing, in accordance with IEEE 344. The program included demonstration of startup and shutdown capabilities, as well as no-load operability during dynamic loading conditions.

3.9.2.2.1.6.10 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.

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 90% of allowable.

3.9.2.2.1.6.11 RHR Heat Exchangers

A dynamic analysis is performed to verify that the RHR heat exchanger will withstand dynamic loadings in accordance with its seismic classification. Dynamic testing is an impractical method to verify the adequacy of equipment when predictable dynamic loads can be determined by analysis.

A three-dimensional finite element model of the RHR heat exchanger and its supports is developed and analyzed using the response spectrum method to verify that the heat exchanger can withstand seismic and hydrodynamic loads. The same model is statically analyzed to evaluate the effects of the external piping loads and dead weight to ensure that the nozzle load criteria and stress limits are met. Critical location stresses are evaluated and found to be lower than the corresponding allowable values.

3.9.2.2.1.6.12 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 for Safe Shutdown Earthquake (SSE) are 1.75g horizontal and 1.75g vertical. The Standby Liquid Control Tank has been dynamically qualified by analysis for:

a. stresses in the tank bearing plate;

b. bolt stresses;

c. sloshing loads imposed at the natural frequency 0.58 hertz;

d. minimum wall thickness; and

e. buckling.

The results confirm that the stresses at the investigated locations are below the allowables.

3.9.2.2.1.6.13 Main Steam Isolation Valves

The main steam isolation valves are qualified for operability by analysis and test. Subsection 3.9.3.2.1.4 outlines the operability assurance program.

3.9.2.2.1.6.14 Main Steam Safety/Relief Valves

The main steam safety/relief valves are dynamically qualified for operability by analysis and test. Subsection 3.9.3.2.1.4 outlines the qualification and operability assurance program.

3.9.2.2.1.6.15 Standby Liquid Control Valve (Explosive Valve)

The standby liquid control valve has been generically qualified to IEEE 344-1975 for seismic and hydrodynamic loads. The generic qualification test demonstrated the absence of natural frequency in the frequency range of the input response spectra and the ability to remain operable after the application of horizontal and vertical dynamic loads in excess of the required response spectra (RRS).

3.9.2.2.2 Seismic Qualification of Safety-Related Non-NSSS (Balance-of-Plant) Mechanical Equipment

This subsection describes the criteria for qualification of nuclear safety-related mechanical equipment and equipment supports. The purpose of qualification is to demonstrate the ability of all such equipment to perform its safety-related function during and after a postulated seismic occurrence of magnitude up to and including the plant-defined SSE.

Attachment A3.9 discusses the BOP considerations for other dynamic loads.

3.9.2.2.2.1 Seismic Qualification Criteria

Seismic qualification of mechanical equipment is accomplished by tests and/or analysis. The selection of testing and/or analysis for a particular piece of equipment is based on the following considerations:

a. If assurance of structural integrity alone can assure the design intended function of the component, then an analysis approach is used for qualification.

b. For components containing mechanisms that must change or maintain a position in order to perform their design intended function, a dynamic testing approach or a combination of testing and/or analysis is performed.

Qualification by testing for active mechanical equipment is discussed in Subsection 3.9.3.2. In addition, qualification by testing of electrical equipment is discussed in Section 3.10.

3.9.2.2.2.2 Seismic Qualification by Analysis

Equipment qualified by an analysis approach provides assurance that the stresses for components designed according to the ASME Boiler & Pressure Vessel Code, Section III, are less than the allowable stress values at each critical section under consideration. The dynamic analysis is also performed to cover the relevant frequency range using a time-history or response spectrum approach. Furthermore, deflections are calculated to assure proper operation of the component.

Methods of Seismic Analysis are discussed in Subsection 3.7.3.5.

3.9.2.2.2.3 Seismic Qualification by Testing

Equipment testing is based on an actual prototype test. The fixture design simulates the actual service mounting and is therefore representative of the actual equipment mounting while in operation.

When equipment is qualified by test, the response spectrum or the time-history at the point of attachment to the supporting structure is the basis for determining the test input motion. The input motion is applied to one vertical and one horizontal axis simultaneously. However, if the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction, and vice versa, the input motion is 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 is such that purely rectilinear resultant input is avoided.

When random vibration input is used, the actual input motion envelopes the specified input motion at all frequencies. However, single-frequency input, such as sine beats, is allowed provided one of the following conditions is met:

a. The characteristics of the required input motion are dominated by one frequency.

b. The anticipated response of the equipment is adequately represented by one mode.

c. The input has sufficient intensity and duration to excite all modes to the required magnitude, such that the test response spectrum envelopes the corresponding required response spectra at all frequencies.

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 during normal reactor operation and from anticipated operational transients.

In general, the vibration forcing functions for operational flow transients and steady-state conditions are not predetermined by detailed analysis. Special analyses of the response signals measured for reactor internals of many similar designs are performed to obtain the parameters which determine the amplitude and modal contributions in the vibration responses. These studies provide useful predictive information for extrapolating the results from tests of components with similar designs.

The dynamic modal analyses also form the basis for interpretation of the preoperational and initial startup test results (Subsection 3.9.2.4). Modal stresses are calculated and relationships are obtained between sensor response amplitudes and peak component stresses for each of the lower normal modes. The allowable amplitude in each mode is that which produces a peak stress amplitude of $\pm$ 10,000 psi.

The basis for $\pm$ 10,000 psi peak stress amplitude is the ASME Code Section III, Appendix I, Table I-9.2, "Design Fatigue Curves for Austenitic Steels...," which gives a low S_a value of approximately 25 ksi for cycles greater than 10^6. General Electric has conservatively limited the peak stress amplitude to 10 ksi to prevent fatigue failure.

The following analysis steps provide details of how the acceptance criteria for sustained vibration stress is applied based upon the calculated natural vibration modal displacements, stresses and frequencies and the observed predominant mode in a test. These steps are identified in a Licensing Topical Report NEDE-24057 (Class I), "Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants", November, 1977. This report was provided to the NRC for Susquehanna SER; the procedure and methodology in the report are in detail, provide the specific examples and date, and apply generically to Clinton reactor internals.

The maximum allowable peak stress amplitude for sustained vibration stress has been specified as 10,000 psi for BWR internals of austenitic stainless steel. This is more conservative than the current ASME Section III allowable stress of 26,000 psi for cycles in excess of 10^6.

To apply this criterion, a dynamic analysis is performed to relate peak stresses to the measured strains or displacements at sensor locations. The steps in this analysis are as follows:

1. Mathematical models are developed using finite element computer codes. The model for the composite structure, including the fuel, shroud, steam separators, reactor pressure vessel (RPV), and control rod drive (CRD) guide tubes is the seismic analysis model for these components. Separate models of the jet pump and feedwater spargers are required.

2. Natural vibration modal displacements, stresses, and frequencies are calculated for each of the lower modes.

3. The location of highest peak stress is identified, and the modal strains and displacements at sensor locations are determined relative to the peak stress on a normalized basis, such that the highest peak stress in each mode is 20,000 psi. This is the allowable stress range, twice the allowable amplitude.

4. The resulting table of strains and displacements for each natural vibration mode and frequency is the criteria used for evaluation of test results.

In applying the criteria, the natural mode which best approximately the observed mode is determined by considering relative amplitudes at different sensor locations on the structure, and also by comparison on observed to calculated frequencies. The stress comparison is then made on the basis of the sensors which are most sensitive to vibration in this mode. Subsection 3.9.2.3.1 of the CPS-USAR provides an example of strain and stress determination in jet pumps. (Q&R MEB (DSER) 68)

3.9.2.3.1 Jet Pumps, Core Support, Steam Separators, and LPCI Coupling

The magnitude of the jet reaction loads applied to the reactor internal structures caused by acceleration and declaration of the flow under normal and upset conditions is negligible compared to the differential pressure loads, and generally need not be considered. Jet reaction loads that require consideration are those associated with the jet pump assembly and riser, and within the steam separator itself. The upward jet reaction loads on the separator assembly are canceled by the downward jet impingement loads at the upper surface of the shroud head dome.

Vibratory loads are continuously applied during normal operation, and the stresses are limited to ± 10,000 psi to prevent fatigue failure. Prediction of vibration amplitudes, mode shapes, and frequencies for normal reactor operations is based on statistical extrapolation of actual measured results on the same or similar components in reactors now in operation. Such predictions from the BWR/4 and 5 experience are made during component design stage. Component design adequacy for flow induced vibration is confirmed through actual in-reactor measurements of the prototype reactor. Kuosheng will be the first BWR/6, 218-inch size to become operational, and is the prototype for Clinton Power Station.

In order to evaluate the dynamic response of the jet pumps, two locations were chosen for monitoring on jet pumps in the prototype plant. These locations are the riser brace and the diffuser of the jet pump. The reasons for selecting these positions were sensitivity and accessibility. Knowing the strain response at these gauge locations, the stresses at other locations can be predicted as well as the mode of vibration, response frequency, and displacement. These values are compared to analytical criteria and thus their acceptability is evaluated.

The load due to cross flow from the jet pumps to the peripheral control rod guide tubes is 620 pounds on the bottom 1/8 of the guide tube length, 345 pounds on the next higher 1/8 of the guide tube length and 130 pounds on the next 1/4 length of the guide tube.

The stresses produced due to vibratory loads are 375.5 psi and are considered negligible.

The dynamic loads due to flow-induced vibration from the feedwater jet impingement would have no significant effect on the steam separator assembly. Analysis has shown that the impingement feedwater jet velocity is 12 ft/sec, far below the critical velocity of 118 ft/sec. Also, analysis has shown that the excitation frequency of the steam separator skirt is 5.1 hertz, and the natural frequency of the skirt is 50 hertz.

The load due to flow induced vibration will have no effect on the LPCI coupling, since the calculated natural frequency of the coupling is over 50 hertz.

The calculated stresses due to the hydrodynamic forces during LPCI operation are small and considered negligible when compared to the design allowable stresses. Locations for which calculations were made include the weld joints, elbows, and rings.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Vibration measurement and inspection programs were conducted for the prototype 218 inch size BWR/6 reactor at the Kuosheng Unit-1 plant in Taiwan during preoperational and start-up testing. (The programs' methods and conclusions are reported in NEDE 22146 (Class III), "Kuosheng-1 Reactor Internals Vibration Measurements", July 1982). The programs were in accordance with the guidelines of Regulatory Guide 1.20 and were conducted in three phases:

a. Preoperational tests prior to fuel loading. Steady-state test conditions included balanced (two-pump) recirculation system operation and unbalanced (single-pump) operation, over the full flow range. Transient flow conditions included single and two-pump trips from rated flow. The test duration subjected major components to a minimum of 10^6 cycles of vibration at the anticipated dominant response frequency and at the maximum response amplitudes. Vibration measurements were obtained during the tests, and a thorough visual inspection of internals was conducted before and after.

b. Precritical testing with fuel. This vibration measurement phase was conducted with the reactor assembly complete but prior to reactor criticality. Flow conditions included balanced, unbalanced, and transient conditions as for the first phase. The purpose of this phase was to verify the anticipated effect of the fuel on the vibration responses of internals. Previous vibration measurements in BWRs have shown that the fuel adds damping and reduces vibration amplitudes of major internal structures (NEDE-24057-P-A (Class III) and NEDE-24057A (Class I), "Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants," April 1981). Thus, the first phase (without fuel) was a conservative evaluation of the vibration levels of these structures.

c. Initial startup testing. Vibration measurements were made during reactor startup at conditions up to 100% rated flow and power. Balanced, unbalanced, and transient conditions of recirculation system operation were evaluated. The primary purpose of this phase was to verify the anticipated effect of two-phase flow on the vibration response of internals. Previous vibration measurements in BWRs have shown that the effect of the two-phase flow was to broaden the frequency-response spectrum and diminish the maximum response amplitude of the shroud and core support structures (NEDE-24057-P-A (Class III) and NEDE-24057A (Class I), "Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants," April 1981).

Strain gauges, displacement sensors (linear variable transformers), and accelerometers were used. Accelerometers were provided with double integration signal conditioning to give a displacement output. The sensor locations were as follows:

a. The upper bolt guide-ring near the top of the separator assembly.

b. The lower shroud wall below the shroud head flange.

c. The shroud just below the shroud head flange.

d. The high pressure core spray line.

e. The fuel channels.

f. The jet pump riser braces.

g. The jet pump diffusers.

h. The low pressure core injection coupling ring.

i. The control rod guide tubes.

j. The incore guide tubes.

k. The LPRM/IRM tubes below the bottom head.

In the Kuosheng prototype plant vibration measurements, only dynamic components of strain or displacement were recorded. Data were recorded on magnetic tape, and provisions were made for selective on-line analyses to verify the overall quality and level of the data. Interpretation of the data required identification of the dominant vibration modes of each component using frequency, phase, and amplitude information from the analyses. Comparisons of measured vibration amplitudes with predicted and allowable amplitudes were made on the basis of the analytically obtained normal mode which best approximate the observed mode.

Analyses of these data showed that all vibrations were within the established criteria.

Reactor internals for Clinton 1 are the same design as Kuosheng Unit-1. Therefore, Kuosheng is a valid prototype per the provisions of Reg. Guide 1.20. CPS internals will be inspected in accordance with the requirements of Regulatory Guide 1.20, Rev. 2, Paragraph 3.1.3, for non-prototypes. Preoperational flow tests will be conducted at the same steady-state conditions and for the same durations as at Kuosheng.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces, a comparison will be 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 will be determined from a 12-node vertical dynamic model of the RPV and internals.

Besides the real masses of the RPV and core support structures, the water inside the RPV will be accounted for.

The accident analysis method is described in Subsection 3.9.5.2.

The time varying pressures are applied to the dynamic model of the reactor internals described above. Except for the nature and locations of the forcing functions and the dynamic model, the dynamic analysis method is identical to that described for seismic analysis and is detailed in Subsection 3.7.2.1. These dynamic loads will be combined with other dynamic loads (including seismic and hydrodynamic), all acting in the same direction, by the square root of the sum of the squares (SRSS) method. This resultant force will then be combined with other steady-state and static loads on an absolute sum basis to determine the design load in a given direction. This analysis is summarized in Table 3.9-2(b).

3.9.2.6 Correlations of Reactor Internals Vibration Tests With the Analytical Results

Prior to initiation of the instrumented vibration measurement program for the prototype plant, extensive dynamic analyses of the reactor and internals are performed (Kuosheng 1 is the prototype plant). The results of the analyses are used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test has been analyzed in detail. The results of the data analysis, vibration amplitudes, and natural frequencies and mode shapes were then compared to those obtained from the theoretical analysis.

Such comparisons provide insight into the dynamic behavior of the reactor internals. The additional knowledge gained from previous vibration tests has been utilized in the generation of the dynamic models for seismic and LOCA analyses for this plant. The models used for this plant are similar to those used for the vibration analysis of earlier prototype BWR plants.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures; and HVAC Ductwork and Duct Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

This section delineates the criteria for selection and definition of design limits and loading combinations associated with normal operation, postulated accidents, and specified seismic and hydrodynamic events for the design of safety-related ASME code components (except containment components which are discussed in Section 3.8).

This section also lists the major ASME Class 1, 2, and 3 pressure parts and associated equipment on a component-by-component basis and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. Design transients for ASME Class 2 equipment are covered in Subsection 3.9.1.1. Seismic related loads are discussed in Subsection 3.9.2.2 and Section 3.7.

Table 3.9-2 is the major part of this section; it presents the loading combination, analytical methods (by reference or example) and also the calculated stress or other design values for the most critical areas in the design of each component, applicable to all ASME Code Class 1, 2, and 3 components, component supports, and core support structures. These values are also compared to applicable code allowables.

3.9.3.1.1 Plant Conditions

All events that the plant might credibly experience during a reactor year are evaluated to establish a design basis for plant equipment. These events are divided into four plant conditions. The plant conditions described in the following paragraphs are based on event probability (i.e., frequency of occurrence) and correlated design conditions defined in the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

3.9.3.1.1.1 Normal Condition

Normal conditions are any conditions in the course of System startup, operation in the design power range, normal hot standby (with condenser available), and System shutdown other than Upset, Emergency, Faulted, or Testing.

3.9.3.1.1.2 Upset Condition

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 any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power, or an operating basis earthquake. Hot standby with the main condenser isolated is an Upset Condition.

3.9.3.1.1.3 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.

3.9.3.1.1.4 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 any pipe break plus the seismic motion associated with a safe shutdown earthquake plus a loss of offsite power, or the safe shutdown earthquake.

3.9.3.1.1.5 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 Encounter Probability Per Reactor Year
Normal (planned)	1.0
Upset (moderate probability)	1.0 > P > 10-2
Emergency (low probability)	10-2 > P > 10-4
Faulted (extremely low probability)	10-4 > P > 10-6

3.9.3.1.1.6 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 inhibit 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.

3.9.3.1.1.7 Compliance with Regulatory Guide 1.48

Compliance with Regulatory Guide 1.48 is shown in Table 3.9-4.

3.9.3.1.1.8 Reactor Pressure Vessel Assembly

The reactor vessel assembly consists of the reactor pressure vessel support skirt, and shroud support.

The reactor pressure vessel and shroud support are constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code. The shroud support consists of the shroud support plate and the shroud support cylinder and its legs. The reactor pressure vessel assembly components are classified as ASME Class I. Complete stress reports on these components have been prepared in accordance with ASME requirements. Table 3.9-2(a) summarizes the loading conditions, calculated stresses, and allowable stresses. The stress analysis performed on the reactor vessel, including the faulted conditions, were completed using elastic methods. The shroud support was also evaluated using elastic conditions except as noted in Subsection 3.9.1.4.3.

3.9.3.1.1.9 Main Steam Piping

The main steam piping discussed in this subsection includes that piping extending from the reactor pressure vessel to the outboard main steam isolation valve. This piping is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB-3600. The load combinations, stress criteria and calculated and allowable stresses are shown in Table 3.9-2(d).

The rules contained in Appendix F of ASME Section III are used in evaluating faulted loading conditions, independently of all other design and operating conditions. Stresses are evaluated on an elastic basis in accordance with F-1360.

3.9.3.1.1.10 Recirculation Loop Piping

The recirculation system piping which is bounded by the reactor pressure vessel nozzles is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB-3600. The load combinations, stress criteria, calculated stresses and allowables are shown in Table 3.9-2(e). The rules contained in Appendix F of ASME Section III are used in evaluating faulted loading conditions, independently of all other design and operating conditions. Stresses are evaluated on an elastic basis in accordance with F-1360.

3.9.3.1.1.11 Recirculation System Valves

The recirculation system flow control and suction and discharge gate valves are designed in accordance with the ASME boiler and Pressure Vessel Code, Section III, Class I, Subsection NB, paragraph 3500. These valves are not required to operate under the safe shutdown earthquake. Loading combinations and other stress analysis information are presented in Tables 3.9-2(f) and 3.9-2(j).

3.9.3.1.1.12 Recirculation Pump

The recirculation pumps are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. These pumps are not required to operate during the safe shutdown earthquake. The loading combinations and other stress information are presented in Table 3.9-2(i).

3.9.3.1.1.13 Standby Liquid Control (SLC) Tank

The standby liquid control tank is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III.

The ASME Code, Subsection ND allowable stress limits for the normal and upset category are l.0S for general membrane and 1.5S for bending (local membrane). The allowable stress limits for the faulted category are 120% of these values, i.e., 1.2S for general membrane and 1.8S for bending (local membrane).

A summary of the design calculations and criteria used is shown in Table 3.9-2(m).

3.9.3.1.1.14 Residual Heat Removal Heat Exchangers

The heat exchangers are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 2 (Subsection NC) for the shell side, and Class 3 (Subsection ND) for the tube side. The stress analysis methods, calculated stresses, and allowable limits for the RHR heat exchangers are shown in Table 3.9-2(o). Heat exchanger design is also discussed in Subsection 3.9.2.2.1.6.11.

3.9.3.1.1.15 RCIC Turbine

Although not under the jurisdiction of the ASME Code, the RCIC turbine has been designed and fabricated following the basic guidelines for an ASME Code, Section III Class 2 component.

Operating conditions for the RCIC turbine include:

a. Surveillance Testing - Quarterly operation with reactor pressure at 1000 psia, nominal, and saturated temperature, turbine exhaust pressure at 25 psia, peak, and saturated temperature.

b. Auto-Startup - 30 cycles per year with reactor pressure at 1150 psia, nominal, and saturated temperature, turbine exhaust pressure at 25 psia, peak, and saturated temperature.

Design conditions for the RCIC turbine include:

a. Turbine Inlet - 1250 psig at saturated temperature.

b. Turbine Exhaust - 165 psig at saturated temperature.

Table 3.9-2(q) contains a summary for the calculated and allowable loads for the RCIC turbine components.

3.9.3.1.1.16 RCIC Pump

The RCIC pump has been designed and fabricated to the requirements for an ASME Code Section III Class 2 component.

Operating conditions for the RCIC pump are surveillance testing in conjunction with the RCIC turbine. A quarterly operation test is performed where the RCIC pump takes condensate from the aboveground storage tank and at design flow discharges condensate back to the aboveground storage tank via a closed test loop.

Design conditions for the RCIC pump include:

a. Required NPSH - 21 feet.

b. Total head - High speed, 2980 feet; Low speed, 610 feet.

c. Constant flow rate - 625 gpm.

d. Normal ambient operating temperature - 60° F to 100° F.

e. Normal plus Upset conditions which control the pump design include:

Design pressure	1525 psig
Design temperature	40° F- 140° F
Operating-basis earthquake	2/3 of SSE

Table 3.9-2(r) summarizes the analysis and allowable limits for the RCIC pumps.

3.9.3.1.1.17 ECCS Pumps

The RHR, LPCS, and HPCS pumps are designed and fabricated in accordance with the requirements of ASME Code Section III. The design conditions are as follows:

	RHR	LPCS	HPCS
Design pressure:			
Suction	215 psig	115 psig	115 psig
Discharge	500 psig	600 psig	1575 psig
Design Temperature	40-360° F	40-212° F	40-212° F

Table 3.9-2(n) summarizes the design calculations for the ECCS pumps.

3.9.3.1.1.18 Standby Liquid Control Pump

The standby liquid control pump has been designed and fabricated following the requirements for an ASME Code, Section III Class 2 component.

Operating conditions for the SLC pump and motor are functionally tested by pumping demineralized water through a closed test loop. The SLC pump is capable of injecting the net contents of the storage tank into the reactor in not less than 50 minutes and not more than 125 minutes. The pump is capable of injecting flow into the reactor against zero psig up to the initial set point of the reactor relief valves.

Design conditions for the SLC pump include:

a. Flow rate - 43 gpm.

b. Available NPSH, maximum 12.9 psi.

c. Maximum operating discharge pressure 1220 psig.

d. Ambient conditions:

Temperature	70° F - 122° F
Relative Humidity	20% - 95%.

e. Normal plus upset conditions which control the pump design include:

Design pressure	1400 psig
Design temperature	150° F
Operating basis earthquake	2/3 of SSE

f. Faulted or emergency conditions include:

Design pressure	1400 psig
Design temperature	150° F
Safe shutdown earthquake	Horizontal 1.75g
	Vertical 1.75g

A summary of the design calculations for the standby liquid control pump components is contained in Table 3.9-2(l).

3.9.3.1.1.19 Main Steam Isolation and Safety/Relief Valves

These valves are designed in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section III, Subsection NB-3500, Safety Class 1.

Load combinations, analytical methods, calculated stresses, and allowable limits are shown for the safety/relief and main steam isolation valves in Table 3.9-2(g) and 3.9-2(h) respectively.

3.9.3.1.1.20 Safety Relief Valve Discharge Piping

The relief valve discharge piping discussed in this subsection includes only that piping extending from the relief valve discharge flange to the first downstream anchor. This piping is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection ND-3600, Class 3. The load combination and stress criteria are shown in Table 3.9-

2(k). The rules contained in the ASME Code Case 1606-1 are used in evaluating faulted loading conditions, independently of all other design and operating conditions.

Each of the 16 SRV discharge lines is rigidly restrained to the drywell wall by a support connected immediately above the quencher inlet. This support consists of a collar-type attachment welded to the piping, as shown in Figure 3.9-13. As per Code Subsection ND requirements, localized bending stresses due to mechanical and thermal expansion loading have been considered for the piping at the attachment location. As shown in Calculation 032182, an envelope of finite element analyses of these connections has been generated. The local stresses from the attachments were added to the pipes' nominal bending and pressure stresses. A summary of maximum pipe stresses is shown in Calculation 032182.

Fatigue concerns for SRV discharge piping were raised by the NRC for BWR Mark I and Mark II plants. Generic programs have addressed these concerns, and have shown acceptable fatigue results. Mark II fatigue results were reviewed with the ACRS on August 7, 1981. The primary fatigue concern was for the portion of the SRV piping in the wetwell airspace, where a fatigue crack could result in steam bypassing the suppression pool. This concern does not apply to the location in question on the Clinton SRV piping, which is submerged in the pool. Also, due to the use of a low-low setpoint logic, the number of design SRV actuations (hence thermal and mechanical load cycles) for Clinton typically will be lower than the Mark I's and Mark II's.

As shown in Calculation 032182, there is a large margin for Equation 10 and 11 stresses. Although not required, thermal gradient stresses if added to Equations 10 and 11 would not create a fatigue problem. Also, a higher allowable stress for these equations could be justified per Code Case N-318, since the number of stress cycles would be less than 7000.

Therefore, per paragraph ND-3645 of the Code, these attachments to the SRV piping were conservatively designed and are well within ND-3650 stress allowables. (Q&R 210.01)

General Electric Company analyzed a total of nine critical locations (four locations are shown on Figure B3.9-1) on the X-Quencher in order to assure that the quencher device meets the requirements of Subsection NB-3600 of ASME Code.

The analysis confirmed that the X-Quencher has been adequately designed to withstand all normal and upset condition loads including localized bending stresses and thermal gradients. See Attachment B3.9 for the General Electric Company "Report on NRC Question About Thermal Gradient Stresses for Clinton X-Quencher." (Q&R 210.02)

3.9.3.1.1.21 <u>Reactor Water Cleanup (RWCU) System</u>

The RWCU pump and heat exchangers are not part of a safety system and not designed to Seismic Category I requirements.

The ASME Code Section III, Class 3 requirements are used as guides in designing the RWCU pump and heat exchangers.

Table 3.9-2(p) shows the calculate stress values and allowable stress limits for the pump.

Table 3.2-1 indicates that the RWCU Heat Exchangers are nonseismic Category I equipment. The acceleration of 0.2g horizontal is used for the static seismic analysis. (MEB (DSER) 72)

3.9.3.1.2 Loading Combinations, Design Transients, and Stress Limits (Non-NSSS)

This section discusses 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 Class MC components) which are discussed in Section 3.8.

Seismic-related loads and seismic qualification by testing and analysis for safety-related mechanical equipment are discussed in Subsection 3.9.2.

ASME Code Seismic Category I fluid system components and supports are required to be designed in accordance with rules and methods specified in the ASME Code. The design stress limits of the ASME Code are selected to ensure the pressure-retaining integrity of safety class equipment. The ASME Code requirements are supplemented by additional requirements in Regulatory Guide 1.48.

The combinations of design loadings are categorized with respect to plant operating conditions/service level limits (stress limits) specified for the evaluation of each ASME Code constructed item in Attachment A3.9.

Class 1 piping meets the criteria based on ASME Code, Section III and 10 CFR 50.55a - Section (d). Class 2 and 3 piping meets the criteria of ASME Code, Section III.

Subsection 3.9.1.1.13 includes the design transients for ASME Code Class 1, 2, and 3 components, CS structures, and component supports.

A description of the computer programs used in the analysis of ASME Code Class 1 components is given in Subsection 3.9.1.2.6.

Experimental methods and design for faulted loads applicable to these components are identified in Subsections 3.9.1.3.4 and 3.9.1.4.12, respectively.

Inelastic methods of analysis are not the preferred methods, but these methods may be used in those cases where it is deemed desirable and appropriate to permit significant inelastic response. In these cases, the system or subsystem analysis performed is modified to include inelastic strain compatibility in the regions on the components and component supports at which significant inelastic response is permitted. These analyses are performed using the NONLIN2, ANSYS4 or PIPERUP Program as described in Appendix C (item C.24) and Subsection 3.9.1.2.6, respectively.

3.9.3.2 Pump and Valve Operability Assurance

3.9.3.2.1 NSSS Active ASME Code Pumps and Valves

The active pumps and valves are listed in Table 3.9-6.

Active mechanical equipment classified as Seismic Category I is designed to perform its function during the life of the plant under postulated plant conditions. Equipment with faulted condition functional requirements include "active" pumps and valves in fluid systems such as the residual heat removal system and the core spray system. (Active equipment must perform a mechanical motion during the course of accomplishing a safety function).

Safety-related valves are qualified by type test or prototype test and analysis. Operability is assured by satisfying the requirements of the following programs.

3.9.3.2.1.1 ECCS Pumps

All active ECCS pumps are qualified for operability by first being subjected to rigid tests before and after installation in the plant. The in-shop tests include (1) hydrostatic tests of pressure-retaining parts in accordance with ASME Section III, (2) seal leakage tests, (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. Also monitored during these operating tests are bearing temperatures (except water cooled bearings) and vibration levels. Both are 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 in-service inspection and operation. These tests demonstrate reliability of the pump for the design life of the plant.

In addition to these tests, the safety-related active pumps are analyzed for operability during a faulted condition by assuring that (1) the pump will not be damaged during the faulted event, and (2) the pump will continue operating despite the SSE faulted loads.

3.9.3.2.1.1.1 Analysis of Loading, Stress, and Acceleration Conditions

In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are limited to the material elastic limit, as indicated in Subsection 3.9.3.1 and Table 3.9-2. The average membrane stress (sm) for the faulted conditions loads is maintained at 1.2S, or approximately 0.75 sy (sy = yield stress) and the maximum stress in local fibers (sm + bending stress sb) is limited to 1.8S, or approximately 1.1 sy. The maximum nozzle loads during a faulted event is also considered in an analysis of the pump supports to assure that a system misalignment cannot occur.

Performing these analyses with the conservative loads stated and with the restrictive stress limits of Table 3.9-2 as allowables, will assure that critical parts of the pump will not be damaged during the faulted condition and that, therefore, the reliability of the pump for postfaulted condition operation will not be impaired.

A dynamic analysis is made to determine the seismic load from the applicable floor response spectra. Analysis is made to check that faulted condition nozzle loads and seismic accelerations will not impair the operability of the pumps during or following the faulted condition. Components of the pump, when having a natural frequency above 33 hertz, are considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas.

If the natural frequency is found to be below 33 hertz, an analysis will be performed to determine the amplified input accelerations necessary to perform the static analysis. The adjusted accelerations will be determined using the same conservatisms contained in the horizontal and vertical accelerations used for "rigid" structures. The static analysis will be performed using the adjusted accelerations; the stress limits stated in Table 3.9-2 must still be satisfied.

3.9.3.2.1.1.2 Pump Operation During and Following Faulted Condition Loading

Active pump/motor rotor combinations are designed to rotate at a constant speed under all conditions. Motors are designed to withstand short period of severe overload. The high rotary inertia in the operating pump rotor, and the nature of the random, short-duration loading characteristics of the seismic event will prevent the rotor from becoming seized. In actuality, the seismic loadings 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 the faulted event and will operate at the design speed.

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 seismic and hydrodynamic loads on the equipment itself. These 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 postfaulted 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 post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.9.3.2.1.2 SLC Pump and Motor Assembly and RCIC Pump Assembly

These equipment assemblies are small, compact, rigid assemblies, with natural frequencies well above 33 hertz. With this fact verified, each equipment assembly has been seismically qualified via static analysis only. This static qualification verifies operability under seismic conditions, and assures structural loading stresses within Code limitations.

3.9.3.2.1.3 ECCS Motors

Qualification of the Class IE motors used for ECCS is in compliance with IEEE 323. The qualification of all motor sizes is based on completion of a type test, followed up with review and comparison of design and material details and seismic analysis of production units, ranging from 600 to 3500 Bhp, with the motor used in the type test. All manufacturing, inspection, and routine tests by motor manufacturer on production units are performed on the test motor.

The type test has been performed on a 1250-hp vertical motor in accordance with IEEE 323 first simulating normal operation during the design life, then the motor being subjected to a number of seismic 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 for the insulation type used on the ECCS motors. The amount of aging equaled the total estimated operation days at 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 and induced vibration effect on the insulation system has been simulated by 1.5g horizontal vibration acceleration at current frequency for 1-hour duration.

d. The dynamic load deflection analysis on the rotor shaft, performed to ensure adequate rotation clearance, has been verified by static loading and deflection of the rotor for the type test motor.

e. Dynamic load aging and testing has been performed on a biaxial test table in accordance with IEEE 344. During this type test the shake table was activated simulating the maximum design limit of the safe shutdown earthquake with motor starts and operation combination as may possibly occur during a plant life.

f. An environmental test simulating a LOCA condition with 100 days duration time has been performed with the test motor fully loaded, simulating pump operation. The test consisted of startup and six hours operation at 212° F ambient temperature and 100% steam environment. Another startup and operation of the test motor after 1-hour stand-still 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 for the insulation type used on the ECCS motors.

3.9.3.2.1.4 NSSS Active Valves - Qualification Method

The safety-related active valves that are ASME Code qualified are Main Steam Isolation Valves, Safety/Relief Valves, Standby Liquid Control Valves, High Pressure Core Spray Valves, and Control Rod Drive Global Valves. Each of these valves is designed to perform its mechanical motion in conjunction with a design basis accident. Qualification for operability is unique for each valve type; therefore, each method of qualification is detailed individually below.

3.9.3.2.1.4.1 Main Steam Isolation Valve

The MSIV's are evaluated by analysis and test for capability to operate under the design loads that envelop the predicted loads during a design basis accident and safe shutdown earthquake.

The valve body is designed, analyzed, and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class I, Subsection NB requirements. The MSIV's are modeled mathematically in the main steamline system analysis. The loads, amplified accelerations and resonance frequencies of the valves, are determined from the overall steamline analysis. The piping supports (snubbers, rigid restraints, etc.) are located and designed to limit amplified accelerations in the valves.

The valve actuator is operated by the plant air and the MSIV air accumulator piping and air volume downstream and including the accumulator check valves are needed to close the MSIVs in the required time. When the valve is in its open position, the yoke and springs are held in compression. The main steam isolation valve with the actuator is modeled in the piping analysis. The axial force and moment at the body-bonnet centerline are predicted from the analysis for the worst combination of piping loads. These values are assured not to exceed the maximum allowable values determined from a simplified valve/actuator analysis, which uses the design g-coefficient of the static equivalent seismic load.

Safety-related limit switches are Class 1E active equipment, qualified in accordance with the requirements of IEEE 382-1972, IEEE 323-1974, and IEEE 344-1975.

3.9.3.2.1.4.2 Main Steam Safety/Relief Valves

The SRV design is qualified by type test to IEEE 344-1975 for operability during a seismic event. Structure integrity of the configuration during a seismic event is demonstrated by both code (ASME B&PV Code Section III, Class 1) analysis and test.

a. Valve is 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 plus dynamic loading (9.0 g horizontal and 6.0 g vertical) of both valve and connecting pipe, thermal expansion of the connecting pipe, and reaction forces from valve discharge.

b. A production SRV demonstrated operability during a dynamic qualification (shake table) type test with moment and "g" loads applied greater than the equipment's design limit loads.

A mathematical model of this valve is included in the main steamline system analysis as with the MSIV's. This analysis assures the equipment design limits are not exceeded.

3.9.3.2.1.4.3 Standby Liquid Control Valve (Explosive Valve)

The SLC Explosive Valve design is qualified by type test to IEEE 344-1975. The valve body is designed, analyzed, and tested according to ASME Boiler and Pressure Vessel Code, Section III, Class 1.

3.9.3.2.1.4.4 High Pressure Core Spray Valves

The HPCS valve body design, analysis, and testing is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 1 or 2. The Class 1E electrical motor actuator is qualified by type test in accordance with IEEE 382-1972, as discussed in Subsection 3.11. Mathematical models of these valves are included in the HPCS piping subsystem analyses. If the valves are found to be rigid, they are qualified for at least the acceleration values obtained from the piping analysis. If the valves are flexible, the piping analysis acceleration values are amplified by a factor of 1.5 and the valves are qualified for at least these higher acceleration values. The operability of the valve assemblies under seismic loading is verified by static pull testing.

3.9.3.2.1.4.5 Control Rod Drive Globe Valve

The globe valves in the CRD scram discharge volume vent and drain lines are evaluated by analysis and test for operability under the design loads that envelop the predicted loads during a design basis accident and safe shutdown earthquake. The valve body is designed, analyzed, and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 2 requirements. The vendor's analysis results indicate that the valves will withstand a maximum acceleration of 4.5 g horizontal and 3.0 g vertical acting simultaneously with a safe shutdown earthquake. The acceptance criteria for seismic disturbance and operability requirements are as follows:

> Stresses must remain within the limits specified for Upset Conditions under Paragraph NB-3223 of Section III of the ASME Boiler and Pressure Vessel Code. The calculated bolt stresses must be less than the normal prestress. The bolt prestress must be less than the limits specified for Upset Conditions under Paragraph NB-3233 of Section III.

The valve actuator is operated by plant air. However, the valve is designed to be fail-safe and the safety operation of the valve closure does not depend upon the plant air supply or on electrical operation of the controlling solenoid valves. When the valve is in its open position, the yoke springs are held in tension. In the event that the solenoid valves that control the globe valves are de-energized, or the plant air supply is interrupted for any reason, the springs are capable of closing the valve.

3.9.3.2.1.5 NSSS Active Valves - Qualification Test Results

3.9.3.2.1.5.1 Main Steam Isolation Valves

The main steam isolation valve following a downstream line break was demonstrated by the "static line test" as defined in the report APED-5750 (March 1969). The test specimen was a 20-inch valve of a design representative of the MSIV's. Operability during seismic accelerations is addressed in Subsection 3.9.2.2.1.

3.9.3.2.1.5.2 Safety/Relief Valves

The safety/relief valve (SRV) has been qualified by type test to both IEEE 323 and IEEE 344. The environmental testing of the sensitive electrical/pneumatic component demonstrated operability after radiation aging (30 x 10^6 rads), thermal aging (392° F for 24 hours), and mechanical aging (1,000 cycles with a specified load). The SRV seismic testing demonstrated operability during the seismic event, and also the absence of natural frequencies below 33 hertz. The SRV was seismically qualified to response spectra induced at the valve inlet flange: 9 g for the horizontal principal axis, and 6 g for the vertical principal axis, with a concurrent static moment load of 800,000 in-lb on the inlet flange and 600,000 on the outlet flange.

3.9.3.2.1.5.3 Standby Liquid Control Valve (Explosive Valve)

The SLC explosive valve has been qualified by type test to both IEEE 323 and IEEE 344. The environmental testing demonstrated operability of the explosive valve (charge) for an exposure to 100% humidity and a peak temperature of 200 degrees F. The temperature of 200 degrees F represents the peak in-situ temperature condition allowing 15 degrees F margin above 185 degrees F during 100 days of DBA condition. The testing also demonstrated operability of the explosive valve to a radiation exposure of 4.17 x 10E5 rads and thermal aging, representing the required 3.75 years of installed life. The dynamic testing demonstrated operability of the valve for seismic and hydrodynamic events, and indicated that the valve will perform its intended safety function as required. A complete description of the updated test results for the Standby Liquid Control valve is given in Reference 7.

3.9.3.2.1.5.4 High Pressure Core Spray Valves

The environmental testing according to IEEE 382-1972 of a specimen motor actuator (see Subsection 3.11) demonstrated operability after radiation aging (2.0 X 10^7 rads), thermal aging (165° F, 100% relative humidity and 200 hours), and mechanical aging (200 cycles during thermal aging and 1,800 additional cycles at ambient temperature). All test data during the seismic aging part of the qualification test show that radiation, thermal, and mechanical aging have no effect on seismic testing.

The type tests were conducted on the valve actuators for seismic qualification without environmental aging. Engineering evaluation showed that the elements of the internal components are rigid members with closely spaced supports with resonant frequencies much in excess of 33 hertz, and that resonance's below 5 hertz were also not possible. Two HPCS valve actuators were subjected to dynamic testing, including a resonance search. The results of the testing for seismic and hydrodynamic events indicate that the valve will perform its intended safety function as required. A complete description of the updated test results for the high pressure core spray valves is given in Reference 8.

3.9.3.2.1.5.5 Control Rod Drive Globe Valves

The valves have no electrical parts that perform a safety function, thus do not require special environment qualification testing that is required for Class 1E equipment. The type tests on the valve and actuator assemblies were conducted in accordance with IEEE 344-1975. The tests showed no malfunctions; the 1C11-F010/F011 vent and drain valves closed in 2 to 3 seconds and the 1C11-F180/F181 vent and drain valves closed in less than or equal to 30 seconds before, during, and after the tests.

3.9.3.2.2 Pump and Valve Operability Assurance - Non-NSSS Systems

Active mechanical equipment (that which performs mechanical motion during the course of accomplishing a safety function) classified as Seismic Category I is shown to perform its function during the life of the plant under postulated plant conditions. Some equipment with these functional requirements are, pumps and valves in fluid systems such as the residual heat removal system, core spray systems, and the isolation systems.

Operability is ensured by satisfying the requirements of the following programs. However, continued operability is ensured by periodic testing.

3.9.3.2.2.1 Pumps

All active pumps are tabulated in Table 3.9-5 and are qualified for operability by first being subjected to 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 1.25 times the system design pressure for ASME Class 1 pumps and 1.5 times the system design pressure for ASME Class 2 and 3 pumps; and (2) performance tests, while the pump is under operation, to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump/motor parameters. 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.

3.9.3.2.2.1.1 Seismic Analysis of Pumps

In addition to the required testing, the pumps are designed and supplied in accordance with the following specified criteria:

a. In order to ensure that the active pump will not be damaged during the seismic event, the pump manufacturer is required to demonstrate by test or analysis that the lowest natural frequency of the pump is greater than 33 hertz. The pump, when having a natural frequency above 33 hertz, will be considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. The natural frequency of the support is determined and used in conjunction with the applicable relevant seismic response spectra.

   In addition, a static shaft deflection analysis of the rotor is performed. The deflection determined from the static shaft analysis is compared to the allowable rotor clearances.

   In case the natural frequency is found to be below 33 hertz, a dynamic or pseudodynamic analysis is performed to determine the amplified input accelerations necessary to perform the stress analysis. In addition, a static deflection analysis is performed as discussed earlier.

b. Nozzle loads from interconnecting piping systems are considered in the stress analysis of the pumps and their supports.

c. To complete the seismic qualification procedures, the pump motor and all appurtenances vital to the operation of the pump are independently qualified for operation during the maximum seismic event in accordance with IEEE 344 (see Section 3.10). In the analysis, interaction between the pump and motor is considered.

From this, it is concluded that the nuclear safety-related pump/motor assemblies will not be damaged and will continue operating under SSE loadings and will perform their intended functions. These requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

3.9.3.2.2.2 Valves

Safety-related active valves are tabulated in Table 3.9-5 and must perform their mechanical motion in times of an accident. Assurance that these valves will operate during a seismic event is obtained by qualification tests or by using a combination of tests and analyses for all active valves.

The safety-related valves are subjected to a series of tests prior to service and during plant life. Prior to installation, the following tests are performed: a shell hydrostatic test according to ASME Section III code requirements; backseat and main seat leakage tests; functional tests to verify that the valve will open and close within the specified time limits when subjected to the design pressure. Qualification of valve actuators in plant harsh environmental zones for the environmental conditions over the installed life of the valve is performed.

Cold hydro qualification tests, functional qualification tests, and periodic inservice inspections are performed to verify and ensure the functional ability of the valve. These tests and appropriate maintenance ensure operability of the valve for the design life of the plant. The valves are designed using either the standard or the alternate design rules of ASME Section III. On all active valves, an analysis of the extended structure is also performed for static equivalent seismic loads applied at the center of gravity of the extended structure. The maximum stresses and deflections allowed in these analyses show adequate structural integrity for these valves.

3.9.3.2.2.2.1 Qualification of Valve Actuators

Each actuator has been qualified to demonstrate its ability to perform its function under all service and environmental conditions. Motors and electrical appurtenances for air actuators are seismically qualified per IEEE 344 and IEEE 323.

3.9.3.2.2.2.2 Check Valves and Safety/Relief Valves

Valves which are safety-related but can be classified as not having an overhanging structure, such as check valves and safety/relief valves, are considered separately.

Due to the particular simple characteristics of the check valves, they were qualified by a combination of the following tests and analysis:

a. stress analysis including the seismic loads where applicable,

b. in-shop hydrostatic test,

c. in-shop seat leakage test, and

d. periodic in situ valve examination and inspection to ensure the functional capability of the valve.

The safety/relief valves are qualified by the following procedures. In-shop hydrostatic, seat leakage, and performance tests shall be performed. In addition to these tests, periodic in situ valve inspection, as applicable, and periodic valve removal, refurbishment, performance testing and reinstallation are performed to ensure the continued functional capability of the valve. In addition, operability of active valves under seismic loading is demonstrated by performing dynamic tests or static pull tests on representative valves.

Using the conservative methods described above these valves were qualified to perform their design function during and following any postulated event. These methods conservatively simulate the seismic event and ensure that the active valves will perform their safety-related function when necessary.

3.9.3.3 Design and Installation of Pressure Relief Devices

3.9.3.3.1 Safety/Relief Valves With Discharge To The Suppression Pool

Safety/relief valve lift results in a transient that produces momentary unbalanced forces acting on the discharge piping system from opening of the valve until a steady discharge flow is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the discharge piping following the relatively rapid valve opening cause the discharge piping to vibrate.

3.9.3.3.1.1 Main Steam Piping

The analysis of the relief valve discharge transient consists of a stepwise time history solution of the fluid flow equation, to generate a time-history of the fluid properties at numerous locations along the pipe. The fluid transient properties are calculated based on the maximum set pressure specified in the steam system specification and the value of ASME flow rating increased by a factor to account for the conservative method of establishing the rating. Simultaneous discharge of all valves is assumed in the analysis because simultaneous discharge is considered to induce maximum stress in the piping. Reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure-times-area, momentum change, and fluid friction terms. Figure 3.9-6 shows a pipe section load transient typical of that produced by relief valve discharge.

The method of analysis applied to determine piping system response to relief valve operation is time history integration. The forces are applied at locations on the piping system where fluid flow changes direction, thus causing momentary reactions. The resulting loads on the safety/relief valve, the main steam line, and the discharge piping are combined with loads due to other effects as specified in Subsection 3.9.3.1. The Code stress limits corresponding to load combinations classified as normal, upset, emergency and faulted, are applied to the steam and discharge pipe.

3.9.3.3.1.2 Other Piping

The design and evaluation of safety/relief valves and discharge piping other than main steam is the same as outlined in Section 3.9.3.3.2 with appropriate consideration for effects due to the submerged leg of pipe in the suppression pool.

3.9.3.3.2 Design and Installation Details for Mounting of Pressure Relief Devices for BOP Systems

The design and installation criteria applicable to the mounting of the pressure-relieving devices for overpressure protection of ASME Class 1 and 2 system components is as follows:

Pressure vessels are protected by pressure-relieving devices to meet applicable code requirements. Detailed design of relief valve stations includes the consideration of both the local stresses at the header-to-relief valve inlet piping junction and the stresses in the relief valve inlet piping and header.

Forces and moments on the piping resulting from thrust developed by full opening of the relief may be considered in the stress analysis. Dynamic load may be considered in the design. The dynamic load factor is defined as the ratio of the actual (or calculated) deflection under dynamic application of a load to the deflection associated with the static application of the same load.

The design of pressure-relieving devices can be generally grouped in two categories: Open discharge and closed discharge.

a. Open Discharge

An open discharge is characterized by a relief or safety valve discharging to the atmosphere or to a vent stack open to the atmosphere.

Regulatory Position C.1 of Regulatory Guide 1.67 recommends that the magnitude of the reaction force, the anticipated transient behavior, and the basis for their determination should be included in the design specification for the valve. Since all of the above are characteristic of each valve design, it is not a current practice to stipulate these data. Rather, the manufacturer supplies that information after selection of the valve, orifice size, inlet and outlet diameter, etc., to accommodate the flow rates required by the design specification. This manufacturer-supplied information is then used in the design of the safety/relief valve station.

The design of open discharge valve stations includes the following considerations:

1. Stresses in the valve header, the valve inlet piping, and local stresses in the header-to-valve inlet piping junction due to thermal effects, internal pressure, seismic loads, and thrust loads are considered. These stresses are calculated in accordance with the applicable subsections of Section III of the ASME Code.

2. Thrust forces include both pressure and momentum effects.

3. Where more than one safety or relief valve is installed on the same pipe run, valve spacing is as specified in ASME Code Case 1569.

4. Where more than one safety or relief valve is installed on the same pipe run, the sequence of valve openings that induces the maximum stresses is considered as recommended by Regulatory Guide 1.67.

5. The effects of the valve discharge on piping connected to the valve header are considered.

6. The reaction forces and moments used in the stress calculations are the maximum instantaneous value obtained from a dynamic time-history analysis or include the effects of a dynamic load factor. A dynamic load factor of 2.0 is used if a dynamic analysis is not performed. Dynamic load factors used are based on Regulatory Guide 1.67 recommendations.

b. <u>Closed Discharge</u>

A closed discharge system is characterized by piping between the valve and a tank or some other terminal end. The design considerations are outlined in Subsection A3.9.4.1.

The design and evaluation of safety relief valves and discharge piping other than main steam is the same as outlined in Subsection 3.9.3.3.2 with appropriate consideration for the effects due to the submerged leg of the pipe in the suppression pool.

The following ECCS relief valves have discharge lines penetrating containment and terminating below the surface of the suppression pool:

Residual Heat Removal (RHR)	High Pressure Core Spray (HPCS)
1E12-F025A, B, & C	1E22-F035
1E12-F017A, & B	1E22-F039
1E12-F005	1E22-F014
1E12-F030	
1E12-F101	Low Pressure Core Spray (LPCS)
1E12-F055A & B (Shut by Raising Relief Setpoint)	
1E12-F036	1E21-F018
1E12-F112A & B	1E21-F031

Based on the expected fluid conditions for each discharge line, the piping has been physically routed to avoid intermittent low points conductive to water hammer phenomenon. For the attached piping, the dynamic loads, such as thrust and momentum caused by the relief valve opening, are evaluated (where applicable) and include such effects as backpressure caused by submergence of the discharge piping in the suppression pool. Where significant (as outlined in the Piping Design Specifications), the dynamic loads have been included in the piping stress analysis. (Q&R MEB (DSER) 75)

3.9.3.4 Component Supports

3.9.3.4.1 Component Supports (NSSS)

3.9.3.4.1.1 Piping

This subsection applies to supports on piping analyzed by the NSSS vendor. The only piping in this scope is the reactor recirculation loop A and B piping described in Subsection 3.9.3.1.1.10 and the main steam piping described in Subsection 3.9.3.1.1.9.

Piping supports are designed in accordance with Subsection NF of ASME Section III. Supports are either designed by load rating per paragraph NF-3260 or to the stress limits for linear supports per paragraph NF-3231. To avoid buckling in the component supports, Appendices F and XVII of the ASME B&PV code require that the allowable loads shall be limited to two-thirds of the critical buckling loads. The critical buckling loads for Class I component supports in the NSSS scope subjected to faulted loads which are more severe than normal, upset and emergency loads, are determined by the vendor using the methods described in Appendix F of the ASME B&PV Code. 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.

The design criteria and dynamic testing requirements for component supports are as follows:

a. Component Supports

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. All component supports are designed in accordance with the rules of Subsection NF of the Code. For all piping in the GE NSSS scope of supply, the valves which are mounted on the piping do not use valve body or operator for component support attachments.

b. 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 their hot and cold load settings. Hangers provide a specified down travel and up travel in excess of the specified thermal movement.

c. Snubbers

Required Load Capacity and Snubber Location

The entire piping system including valves and suspension system between anchor points is mathematically modeled for complete structural analysis. In the mathematical model, the snubbers are modeled as a spring with a given spring stiffness depending on the snubber size. The analysis determines the forces and moments acting on each component and the forces acting on the snubbers due to all dynamic loading conditions defined in the piping design specification. The design load on snubbers includes those loads caused by seismic forces (operating basis earthquake and safe shutdown earthquake) system anchor movements and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

The snubber location and loading direction are first decided by estimation so that the stresses in the piping system will have acceptable values. The snubber locations and direction are refined by performing the computer analysis on the piping system as described above.

The spring constant required by the suspension design specification for a snubber of given load capacity is compared against the spring constant used in the piping system model. If the spring constants are the same, then the snubber location and load direction have been confirmed. If the spring constants are not in agreement, they are brought in agreement, and the system analysis redone to confirm the snubber loads. This iteration is continued until all snubber load capacities and spring constants are compatible.

Design Specification Requirements

To assure that the required structural and mechanical performance characteristics and product quality are achieved, the following requirements for design and testing are imposed.

1. The snubbers are required by the suspension design specification to be designed in accordance with all of the rules and regulations of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. This

design requirement includes analysis wherein the stresses in the snubber component parts are calculated under normal, upset, emergency and faulted loads. These calculated stresses are then compared against the allowable stresses of the material as given in Appendix I of ASME Section III Code, to make sure that they are below the allowable limit.

2. The snubbers are tested to ensure that they can perform as required during the operating basis earthquake (OBE), the safe shutdown earthquake (SSE), hydrodynamic loads and under anticipated operational transient loads or other mechanical loads associated with the design requirements for the plant. The test requirements include:

a) Snubbers are subjected to force or displacement versus time loading at frequencies within the range of significant modes of the piping system.

b) Displacements are measured to determine the performance characteristics specified.

c) Tests are conducted at various temperatures to ensure operability over the specified range.

d) Peak test loads in both tension and compression will be equal to or higher than the rated load requirements.

e) The snubbers are also tested for various abnormal environment conditions. Upon completion of the above abnormal environmental transient test, the snubber shall be tested dynamically at a frequency with a specified frequency range. The snubber must operate normally during the dynamic test.

Snubber Installation Requirements

An installation instruction manual is required by the suspension design specification. This manual is required to contain instructions for storage, handling, erection and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing, which contains the installation location of the snubber on the pipe and structure, the hot and cold settings, and additional information needed to install the particular snubber.

The suspension design specification requires that snubbers be provided with position indicators to identify the rod position. This indicator facilitates the checking of hot and cold settings of the snubber, as specified in the installation manual, during plant preoperational and startup testing.

Inspection, Testing, Repair and/or Replacement of Snubbers

The suspension design specification requires that the snubber supplier prepare an installation instruction manual. This manual is required to contain complete instructions for the testing, maintenance and repair of the snubber. It also contains inspection points and the period for inspection.

The suspension design specification requires that hydraulic snubbers be equipped with a fluid level indicator so that the level of fluid in the snubber can be ascertained easily.

d. Struts

The design load on struts includes those loads caused by dead weight, thermal expansion, primary dynamic forces, i.e., operating basis earthquake (OBE) and safe shutdown earthquake (SSE), hydrodynamic loads. system anchor displacements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

Struts are designed in accordance with NF-3000 to be capable of carrying the design load for all operating conditions.

3.9.3.4.1.2 Reactor Pressure Vessel (RPV) Support Skirt

The RPV support skirt is designed as an ASME Code Class I plate and shell type component support per the requirements of ASME Boiler and Pressure Vessel Code Section III, Subsection NF. The loading conditions, stress criteria, calculated stresses and the allowable stresses in the critical support areas in Table 3.9-2(a). The stress level margins prove the adequacy of the RPV support skirt.

3.9.3.4.1.3 NSSS Floor Mounted Equipment (Pumps, Heat Exchanger and RCIC Turbine)

The High Pressure Core Spray, Low Pressure Core Spray, Residual Heat Removal, Reactor Core Isolation Cooling, Standby Liquid Control, Reactor Water Cleanup Pumps, Residual Heat Removal, Reactor Water Cleanup Heat Exchangers and RCIC Turbine are all analyzed to verify the adequacy of their support structure under various plant operating conditions. In all cases the stress loads in the critical support areas are within the ASME Code allowables. The loading conditions, stress criteria and the allowable stresses in the critical support areas are summarized in Table 3.9-2 under the respective equipment table.

The stress levels within the ECCS pumps are below the ASME allowable stress and therefore will not cause permanent deformation. Adequate clearance is allowed in the design so that mechanical interference does not exist and has been verified.

The pumps are tested for spindown time (time to coast to a stop) which verifies that mechanical interference does not exist. Pump bearings are designed to limit shaft deflections in the faulted condition. Calculations are contained in the design report. The motor stand is designed to withstand the faulted (plant) stress with insignificant deformation. (Q&R MEB (DSER) 78)

3.9.3.4.1.4 Supports for ASME Code Class 1, 2 and 3 Active Components

ASME Code Class 1, 2 and 3 active components are either pumps or valves. Since valves are supported by piping and not tied to building structures, pipe design criteria govern.

Seismic Category I active pumps supports are qualified for seismic and hydrodynamic loads by testing when the pump supports along with the pumps are fulfilling the following conditions:

a. Simulate actual mounting conditions.

b. Simulate all static and dynamic loadings on the pump.

c. Monitor pump operability during testing.

d. The normal operation of the pump during and after the test indicates that the supports are adequate. Any deflection or deformation of the pump supports which precludes the operability of the pump, is not accepted; and,

e. Supports are inspected for structural integrity after the test. Any cracking or permanent deformation is not accepted.

Seismic and hydrodynamic qualification of component supports by analysis is generally accomplished as follows:

a. Stresses at all support elements and parts such as pumps holddown, and baseplate holddown bolts, pump support pads, pump pedestal, and foundation are checked to be within the allowable limits as specified in ASME Subsection NF.

b. For normal and upset plant conditions, the deflections and deformations of the supports are assured to be within the elastic limits and not exceed the values permitted by the designer based on his design verification tests to ensure the operability of the pumps.

c. For emergency and faulted plant conditions, the deformations must not exceed the values permitted by the designer to ensure that operability of the pumps.

d. For piping not within the GE NSSS Scope of Supply, valves and valve operators are used as component attachment points. Attachments have been made to the NSSS valves as noted in Table 3.9-15. (Q&R MEB (DSER 76).

3.9.3.4.2 Component Supports (Balance of Plant)

3.9.3.4.2.1 Piping

Supports for ASME Class 1, 2, and 3 nuclear piping are designed in accordance with Subsection NF of ASME Section III 1974 including addenda through Summer 1974 with the exception of members/items classified as AISC Supplementary Structural Steel. AISC Supplementary Structural Steel members/items are designed for the same loading combination as the component support items classified as NF. For Operating and Design load combinations, the stresses are limited to those specified in the AISC Specification, Part I. For Emergency and Faulted load combinations, the allowable stresses are limited to 1.6 times those specified by the AISC Specification, Part I, or 0.95 Fy (yield stress), whichever is smaller.

These supports are designed by the following criteria:

Type	Method	Reference
Component Standard Supports	Load Rating	NF-3260

Linear	Analysis	NF-3230, 3260 *
Plate & Shell	Analysis	NF-3220, 3260 *

*Additional requirements per Regulatory Guides 1.124 and 1.130 as defined in Section 1.8 of the CPS-USAR.

The load combinations for the various system operating conditions are taken into account in designing component supports for ASME Code constructed items. Design transient cyclic data are not applicable to piping supports as no fatigue evaluation is necessary to meet the code requirements.

Supports loading include but are not limited to the following:

a. Weight of the component and normal contents under operating and test conditions (including insulation where applicable).

b. Dynamic loads, including loads caused by earthquake SRV actuation, LOCA events and other system design mechanical loads.

c. Restrained thermal expansion.

d. Relative anchor and support movements.

The design loading combinations, including transients and stress and deformation limits, for ASME Code Class 1 reactor coolant pressure boundary supports and internals structures are presented in Subsections 5.4.14 and 3.9.5, respectively. Loadings and stress limits for ASME Code, Class 2 and 3 component supports are discussed in Subsection 3.9.3.1. Active component supports are considered in Subsection 3.9.3.2. The stress limits are per NF-3000 for design, normal, upset, and emergency and ASME Section III, Appendix F, for faulted conditions.

For BOP piping, component supports have been attached to the valve body or yoke to limit acceleration; however, no supports are attached to the valve operators. Attachments have been made to the BOP valves as noted in Table 3.9-16. (Q&R MEB (DSER) 76)

All component supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after installation. Valves with component support attachments are qualified for the appropriate service conditions, including valve operability as required.

The design criteria and dynamic test requirements for component supports are as follows:

a. <u>Hangers (Spring Type)</u>

The design load on hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the design load at both their hot and cold load settings. Hangers provide a specified down travel and up travel in excess of the specified thermal movement.

b. <u>Struts and Rigid Hangers</u>

The design load 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 displacement, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

Struts are designed in accordance with NF-3000 to be capable of carrying the design load for all operating conditions.

c. Snubbers (Mechanical)

The design load on snubbers includes those loads caused by seismic forces (operating-basis earthquake and safe shutdown earthquake), system anchor movements and reaction forces caused by relief valve discharge, turbine stop valve closure, etc., as applicable to the unit's intended function.

The snubbers are designed and load rated in accordance with NF-3000 to be capable of carrying the design load for all operating conditions. Faulted condition design used the criteria outlined in Appendix F of the ASME Code. They are designed to be able to carry the load under normal, upset, emergency, and faulted loading conditions.

Prototype snubbers have been tested dynamically to ensure that they can perform as required in the following manner:

1. The snubber was subjected to a sinusoidal forcing function.
2. The frequency (hertz) of the input force was verified at small increments within the specified range.
3. The resulting relative displacements and corresponding loads across the working components, including end attachments, were recorded.
4. The peak load in both static tension and compression tests was higher than the rated load followed by an operational test.
5. The duration of the tests at each frequency was specified.
6. Snubbers were tested for various normal and abnormal environment conditions, including temperature, radiation, and humidity, followed by operational tests.
7. Details of the operational tests used to verify the capability of the units to perform their intended function are contained in Pacific Scientific Co. qualification test reports as follows:

Test Report Number	Unit
TR 807	PSA-1
TR 808	PSA-3
TR 809	PSA-10

TR 810	PSA-1/4
TR 811	PSA-1/2
TR 812	PSA-35
TR 814	PSA-100

d. Anchors are designed to secure at the desired points of piping in relatively fixed positions. They permit piping to expand and contract freely as directed from the anchor and are structurally capable of withstanding thrusts, moments, and other imposed loads.

BOP Snubbers

Each snubber was alternately loaded in tension and compression to the rated (design, normal, and upset) load (see Table 3.9-17) with the criteria that the total travel of the unit during cyclic loading, including lost motion and deflection, shall not exceed ± 0.060 inch (0.120 inch total). The unit was subjected to load cycling at 100, 75, and 50 percent of the rated load at frequencies of 3, 6, 9, and 12 Hz for the time specified in Table 3.9-17 at each 3-Hz step (6 minutes total). In addition, the unit was subjected to 100 percent rated load between 15 Hz and 33 Hz by applying a single load pulse in both tension and compression.

The faulted load, as shown in Table 3.9-17 was applied for one minute in tension and compression. The snubbers were not tested for emergency loads.

NSSS Snubbers

Two snubbers of each size and each model were tested under upset and faulted loads in the manner described below:

a. Snubbers were tested dynamically to ensure that they could perform as required under upset loading conditions in the following manner:

   1. The snubbers were subjected to a force that varied approximately as the sine wave.

   2. The frequency (Hz) of the input force was in increments of 5 Hz within the range of 3 to 33 Hz.

   3. The test was conducted with the snubber at room temperature and at 200° F.

   4. The peak load in both tension and compression was equal to or higher than the rated load of the snubbers.

   5. The duration of the test at each frequency was 10 seconds or more.

b. Snubbers were tested dynamically to ensure that they could perform as required under emergency and faulted loading conditions in the following manner:

   1. The snubbers were subjected to forces that varied approximately as a sine wave.

2. The test was conducted with the snubber at room temperature.

3. The peak load in both tension and compression was equal to 1.5 times the rated load of the snubbers.

4. The duration of the test was 10 seconds.

The following materials supplement the existing response with regard to the NSSS Snubber design:

Snubbers are qualified for service by General Electric by testing for bleed, lockup rate, drag or friction force and for response to dynamic loading. The dynamic loading test is accomplished by subjecting the snubber to a sinusoidal force that is equal to the rated load of the snubber. The force is applied at frequencies that are at 5Hz increments within the range of 3 Hz to 33 Hz. The dynamic load tests are conducted with the snubber at both room temperature and at 200° F.

The snubbers are modeled as linear elastic springs in the dynamic analysis of the piping system. The vast majority of all dynamic loadings occurs with frequencies ranging form 3 to 33 Hz. By using the results of the dynamic testing, spring constants are calculated. These constants increase with higher frequencies. The average spring constant, including all lost motions (dead band, etc.) of the snubber, is then used by General Electric in the analytical model of the snubber.

In addition to the testing of the snubbers by themselves, General Electric has subjected the CPS safety relief valve piping to safety relief valve discharge while monitoring the piping system for stresses. The safety relief valve discharge creates acoustic waves that propagate through the safety relief valve piping and impose momentary forces on the pipe at each change in direction. The results of this testing of the piping system show a satisfactory correlation between actual stresses and predicted stresses in the pipe. Since the analytical model of the piping system uses the spring constant obtained from the aforementioned snubber test, this correlation serves as a calibration of the snubber spring constant.

Although the frequencies induced on the snubbers have not been measured directly, the vibration frequencies of the piping have been obtained for the main steam line during the Monticello and Hatch 2 tests and the frequencies of the main steam and recirculation piping have been measured during the Caorso test. The frequencies induced on the snubbers may be assumed to be closely related to the piping vibrating frequencies.

Although frequencies were measured from 5 Hz to 50 Hz, the dominant frequencies varied from 12 Hz to 36 Hz.

The tests at Monticello and Hatch 2 measured response of the piping system to the acoustic wave in the SRV discharge piping following SRV opening. These loads are a function of the configuration of the discharge piping and are not related to the configuration of the containment. The Caorso tests measured the responses of the piping system to the acoustic wave in the discharge piping and also measured the responses of the piping to the hydrodynamic load associated with the SRV discharge. Although the Mark II and Mark III containments are different, the Caorso tests show that the structural responses (accelerations) to the hydrodynamic load in the suppression pool was much smaller than the calculated values at design conditions. It is expected that the load for Mark III containment due to SRV will also be much smaller than the calculated values. The frequencies at which the piping responds to the hydrodynamic loads are not expected to differ significantly in the Mark III containment.

The stresses in the main steam branch pipe of a BWR due to safety/relief valve blowdown were measured from an in situ piping system test (Hatch). The test results were compared with analytical results. The calculated stresses for SRV discharge piping response loads were found to be conservative when compared to measured stress values. Table 3.9-18 gives the ratios of measured to calculated stress values. MEB (DSER) ITEM NO. 79)

The design of all supports for non-nuclear piping shall satisfy the requirements of ANSI B31.1.

3.9.3.5 HVAC Ductwork and Duct Support Structures

HVAC ductwork and duct supports are designed using a frequency controlled design approach. By using a frequency controlled design, the ductwork and duct supports are effectively decoupled and each component is designed in the rigid frequency range of the appropriate floor response spectra. The qualification is done independently for the ductwork and the duct supports.

For the ductwork support evaluation it is assumed that all HVAC ductwork and accessory tributory weights are transferred to the supporting structure. The stresses in the ductwork support structures are limited to the AISC Specifications, Part I, for the loading combinations specified in Tables A3.9-6 and A3.9-7 and the Stress Limits shown in Table 3.9-14.

The evaluation of the HVAC ductwork stresses is done for both local and gross effects. The design rules of AISI (Cold-Formed Steel Design Manual - Part 1) are used with the stresses limited to those shown in Table 3.9-14 for the loading combinations specified in Tables A3.9-6 and A3.9-7.

During the NRC site inspection, two vertical duct risers were discovered to contain buckled portions near the top-most supports.

The subsequent investigation revealed that an inadequate duct gauge thickness combined with a non-optimum installation sequence caused the buckled conditions.

The two long vertical duct risers were constructed of 22 gauge material instead of 18 gauge is designed. They are constructed from the top and downward.

Sargent & Lundy evaluated the stress in the two ducts and found that buckling of the 22 gauge material was expected. However, the analysis disclosed that the buckled 22 gauge ductwork would not have failed under design dynamic loads. The calculations also showed that buckling of the specified 18 gauge material under similar construction methods would not occur.

In its review of the design calculation for the two HVAC risers, Sargent & Lundy verified that the ductwork can withstand all normal and seismic forces, including the normal weight loads experienced during the construction of long vertical duct risers.

One of the buckled 22 gauge risers has been replaced with 18 gauge material and the other buckled duct riser has had stiffeners added. Also, all vertical risers are being reviewed for proper material gauge. (Q&R 210.06)

3.9.4 Control Rod Drive System

This plant is equipped with a hydraulic control rod drive system which includes the control rod drive mechanism (CRDM), the hydraulic control unit (HCU), the condensate supply system and the scram discharge volume, and extends to the coupling interface with the control rods.

3.9.4.1 Descriptive Information on CRDS

Descriptive information on the control rod drives as well as the entire control and drive system is contained in Section 4.6.

3.9.4.2 Applicable CRDS Design Specifications

The control rod drive system (CRDS) is designed to meet the functional design criteria as outlined in Section 4.6 and consists of the following:

a. locking piston control rod drive;

b. hydraulic control unit;

c. hydraulic power supply (pumps);

d. interconnecting piping;

e. flow and pressure and isolation valves; and

f. instrumentation and electrical controls.

Those components of the CRD forming part of the primary pressure boundary are designed according to ASME Code Section III, Subsection NB.

The mechanisms which have caused cracking in operating BWRs are understood. A summary discussion of the previously observed problems and the solutions incorporated in the CPS design are presented in the following.

A detailed evaluation of the problems of the feedwater nozzle and sparger is presented in Reference 10. The solution of the feedwater nozzle and sparger cracking problems involves several elements, including material selection and processing, nozzle clad removal, and thermal sleeve and sparger redesign. The following summarizes the problems that have occurred in the nozzle and sparger and shows the solution that eliminates each problem:

PROBLEM	CAUSE	FIX
Sparger arm cracks	Mechanical fatigue	Eliminate/minimize clearance between thermal sleeve and safe end
	Thermal fatigue	Eliminate low flow stratification by use of topmounted elbows
Flow hole cracks	Thermal fatigue	Eliminate separation by use of

		converging nozzles
Nozzle cracks	Thermal fatigue	Eliminate clad, control leakage, protect nozzle with multiple sleeves

The sparger vibration has been attributed to a self-excitation caused by instability of leakage flow through the annular clearance between the thermal sleeve and safe end. Tests have shown that the vibration is eliminated if the clearance is reduced sufficiently or sealed. The solution which has been selected uses a two-stage piston ring seal mounted in the thermal sleeve in conjunction with an interference fit between the sleeve and safe end. This feature is also an essential part of the solution of the nozzle cracking problem and is described later in more detail. Freedom from vibration over a range of conditions has been demonstrated by the tests reported in Reference 10, Section 4.

Sparger arm cracking has also been caused by thermal fatigue, both at the flow holes and adjacent to the tee connection with the thermal sleeve. In both cases, excessive cyclic thermal stresses are caused by the exposure of material in a constrained structure to an unstable boundary between cold feedwater fluid and hot reactor fluid. At low feedwater flow, the presence of exit flow holes at the midplane of the sparger allowed the sparger to be only partially filled with cold fluid. This caused a temperature gradient from the top of the sparger to the bottom, with associated bending stresses which changed directly with changes in the flow gradient. Relocation of the exit flow holes at the top of the sparger allows complete filling of the sparger with the feedwater fluid even at low flow, producing a more stable and homogenous temperature distribution. As shown by the data reported in Reference 10, Section 4.3, stratification has been eliminated over the range of operating flows.

Flow hole cracks occurred partly because the surface of the hole was constrained by the in-plane stiffness of the surrounding sparger material when exposed to the exit flow to reactor coolant gradients, and partly because the gradients themselves were unstable. The instability of the gradients resulted from changing location of the separation point between the cold exit flow and the warmer boundary layer produced by heating of the sparger by reactor fluid. The result was a high-cycle thermal stress around the edge of the hole. This condition is eliminated by the exit flow elbows which have a long enough exit throat to stabilize the flow separation. Also, the thermal stress produced by a given gradient is much less with the exit hole in a cylindrical tube, rather than in what previously would behave locally more as a flat plate. Testing, as reported in Reference 10, Section 4.3, has shown that the high frequency thermal cycling is eliminated by the new design.

In order to allow for removal of the sparger, it is necessary to provide a sealed joint between the nozzle safe end and the thermal sleeve. This seal is achieved by use of a metal piston ring backed up with a coil spring expander. Even if the piston ring seal was leaktight when initially, installed, its long-term sealing ability is unknown. The effects of wear and corrosion on the mating safe end surface would eventually cause leakage to increase to the point where nozzle cracking would initiate. The rate of deterioration in unpredictable, but is expected to be short relative to the life of the pressure vessel. To provide protection against seal failure resulting in nozzle cracking, the second piston ring and the added thermal sleeves have been incorporated in the new design. It has been demonstrated by test that the triple thermal sleeve arrangement prevents the leakage flow causing nozzle cracking. This is the result of the concentric sleeve arrangement channeling leakage away from the nozzle and the fact the second seal is exposed to very low driving pressures, making leakage past it very small.

As mentioned earlier, the cracking of the feedwater nozzles is a two-part process. The crack initiation mechanism as discussed above is the result of self-initiated thermal cycling. If this were the only mechanism present, the cracks would initiate, grow to a depth of approximately 0.25 inch, and arrest. This degree of cracking would be tolerated, but unfortunately there is another mechanism which supports crack growth. This mechanism is the system-induced transients, primarily the startup/shutdown transients. The triple thermal sleeve arrangement also assists in this area because, even with the piston rings leaking, the heat transfer coefficient between the feedwater and the nozzle are reduced to the point where the thermal stresses in the nozzle are not high enough to cause significant crack growth. Analysis presented in Reference 10, Section 4,6, demonstrates this benefit and the benefit of using unclad nozzles.

The cracking of the CRD return nozzles is caused by a mechanism which is very similar to that which caused cracking in the feedwater nozzles, i.e., thermally induced fatigue.

The CRD return flow is always at a low temperature (40° to 140° F). The flow rate is also low and as the fluid passes through the nozzle it mixes with the hot (580° F) reactor coolant. This mixing is turbulent and results in alternating hot and cold cycling on the nozzle wall. The result in high cycle fatigue which initiates cracking. This mechanism has been demonstrated by test. Tests have also demonstrated that lower frequency thermal cycles occur in a stagnant CRD return line nozzle.

The fix for this problem is the elimination of the CRD return flow to the vessel nozzle. It has been shown that the CRD system will operate satisfactorily with the return line cut and capped. This has been demonstrated by tests at Peach Bottom, Fitzpatrick, and other operating BWRs, and is allowed by NUREG 0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Cracking," November 1980.

Stress analysis in keeping with the requirements of the ASME Code Section III will be performed to demonstrate the adequacy of the reactor vessel feedwater nozzle (and sparger) and the CRD return line nozzle cap. Compliance with NUREG 0619 will be demonstrated. (Q&R MEB (DSER) 91)

The quality group classification of the CRD hydraulic system is outlined in Table 3.2-1. The components are designed according to the codes and standards governing the individual quality groups.

Pertinent aspects of the design and qualification of the CRD components are discussed in the following locations: transients in Subsection 3.9.1.1., faulted conditions in Subsection 3.9.1.4, seismic testing in Subsection 3.9.2.2, and loading combinations in Table 3.9-2 (u).

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformation

The ASME Code components of the CRDs and CRD Housings have been evaluated analytically and the design load combinations and stress limits are listed in Tables 3.9-2 (u) and 3.9-2 (v). For the non-code components, experimental testing was used to determine the CRD performance under all possible conditions as described in Subsection 3.9.4.4. Deformation has been compared with allowable and is not a controlling factor based upon the numerous tests performed on the drive.

3.9.4.4 CRD Performance Assurance Program

The CRD of the BWR/6 design has undergone extensive testing under simulated BWR/6 reactor conditions to assure performance and structural integrity. The test program consists of the following:

a. development test,
b. design acceptance test,
c. manufacturing quality control test,
d. production verification test,
e. 1.5x design life test,
f. operational test, and
g. surveillance test.

In addition to the following discussions, test programs a., b., c., f., and g. are further discussed in Subsection 4.6.3.

3.9.4.4.1 Development Test

This test was conducted to examine various drive performance parameters which yielded an optimum design for achieving the design objectives. The control rod drive components were also evaluated for their structural integrity under various simulated reactor conditions.

3.9.4.4.2 Design Acceptance Test

This test was conducted to verify the final design concept using a prototype control rod drive fabricated to resemble production hardware. In this test, the control rod drive was subjected to the entire range of expected operational BWR/6 reactor conditions and postulated abnormal BWR/6 conditions. In addition, the field evaluation of the drive includes the installation and operation of the drive for approximately 1 year, then removal and replacement with a second drive for long-term evaluation.

3.9.4.4.3 Manufacturing Quality Control Test

This test is intended to establish an adequate data base for acceptance testing of production control rod drives. These drives were tested beyond the designed life to establish design and reliability margins.

3.9.4.4.4 Production Verification Test

Four control rod drives are normally picked at random from the production stock each year and subjected to various tests under simulated reactor conditions for a period of one maintenance life cycle (5 years) and 1/6 of the cycles specified in Subsection 3.9.1.1.

This phase of testing was intended to verify the manufacturing procedures. Prior to shipment, each new production drive's performance must satisfy the quality control requirements by test.

The following statement is historical:

Prior to initial reactor startup, all normal drive functions shall be reverified to demonstrate that each drive, as installed in the reactor, satisfies the performance criteria.

The BWR/6 CRD design acceptance test is reported in "Fast Scram Control Rod Drive Qualification Program", NED0-24142.

3.9.4.4.5 1.5 X Design Life Tests

When a significant design change is made to the components of the drive, the drive is subjected to a series of tests equivalent to 1.5 times the life test cycles specified in Subsection 3.9.1.1.

Two CRD's underwent such testing in 1976. upon completion of the test program, these CRD's met or surpassed the minimum specified performance requirements.

3.9.5 Reactor Pressure Vessel Internals

This subsection identifies and discusses the structural and functional integrity of the major reactor pressure vessel internals.

3.9.5.1 Design Arrangements

The core support structures and reactor vessel internals (exclusive of fuel, control rods, and incore nuclear instrumentation) are identified in the following paragraphs:

a. Core Support Structures

1. Shroud – Including shroud stabilizer assemblies.

2. Shroud support cylinder, plate and legs (part of the RPV Core plate, and core plate hardware.

3. Grid (Only that portion below the bottom weld in the cylindrical portion is core support structure. The grid is a part of the top guide assembly).

4. Top guide hardware (studs and nuts between top guide and shroud).

5. Orificed fuel supports (except for orifices which do not support or restrain the core).

6. Peripheral fuel support.

7. CRD Housing (only that portion of the CRD housing that is above the housing to pressure vessel weld).

8. Control rod guide tubes.

b. Reactor Internals

1. Jet Pump assemblies, braces, and instrumentation.

2. Feedwater spargers.*

3. Vessel head spray nozzle.

4. Differential pressure and liquid control lines.

5. In-core instrument guide tubes and stabilizers.*

6. Initial startup neutron sources (when installed).*

7. Surveillance sample holders.*

8. Core spray lines and spargers.

9. In-Core Instrument housings. *

10. LPCI Coupling.

A general assembly drawing of the important reactor components is shown in Figure 3.9-7.

The floodable inner volume of the reactor pressure vessel can be seen in Figure 3.9-8. It is the volume inside the core shroud up to the level of the jet pump suction inlet.

The design arrangement of the reactor internals, such as the jet pumps, steam separators, and guide tube, is such that one end is unrestricted and thus free to expand.

The LPCI couplings incorporate vertically oriented slip fit joints to allow free thermal expansion.

*Non-safety class components

3.9.5.1.1 <u>Core Support Structures and Reactor Vessel Internals</u>

The core support structures consist of those items listed in Subsection 3.9.5.1. These structures form partitions within the reactor vessel, to sustain pressure differentials across the partitions, direct the flow of the coolant water, and laterally locate and support the fuel assemblies. Figure 3.9-8 shows the reactor vessel internal flow paths.

3.9.5.1.1.1 <u>Shroud</u>

The shroud support, shroud, and top guide make up a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus, thus providing a floodable region following a recirculation line break. The volume enclosed by this assembly is characterized by three regions. The upper portion surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide's grid plate below. The central portion of the shroud surrounds the active fuel and forms the longest section of the assembly. This section is bounded at the top by the grid plate and at the bottom by the core plate. The lower portion, surrounding part of the lower plenum, is welded to the reactor pressure vessel shroud support.

3.9.5.1.1.1.1 <u>Shroud Stabllizer Assemblies</u>

Shroud stabilizer assemblies are installed to implement the General Electric (GE) H1/H7 Reactor Pressure Vessel (RPV) core shroud repair design as an alternative repair to the core shroud. The shroud stabilizer assemblies consist of radially acting stabilizers mounted on four vertical mechanically preloaded tie rods to maintain the alignment of the core shroud to the reactor pressure vessel (RPV), and the originally designed reactor flow partitions. The set of stabilizers replaces the structural functions of the top guide/grid and shroud horizontal welds H1 through H7, which are assumed to contain 360 degrees through wall cracks. Each stabilizer assembly consists of a tie rod, an upper and lower stabilizer, an upper support, and other connecting members. The tie rod and upper support provide the vertical load restraint capability from the top of the shroud to the RPV shroud support, as well as positioning the new radial stabilizers. The tie rod preload acts downward on the top surface of the top guide/grid at four equally spaced azimuths. It is reacted by its toggle attachment at the bottom to the shroud support plate. The tie rod and stabilizers are installed with relatively low mechanical preloads, assuring they are held tightly in place and do not vibrate during plant startup. The tie rod preload increases at operating temperature, due to differential expansion between the Alloy X-750 tie rod and the stainless steel shroud. This gives an operating preload sufficient to prevent cracked shroud joints from separating.

The vertical locations of the radial stabilizers are chosen to provide positive positioning for all segments of the shroud and the fuel assemblies, assuming that the shroud welds H1 through H7 contained through wall cracks. The upper stabilizers provide radial restraint against the RPV at the top guide elevation. The upper support provides an attachment feature for the assembly to the top of the shroud, and vertical restraint against separation of the shroud welds. The lower stabilizers act through core plate wedges to provide radial restraint between the shroud at the core support plate elevation and the RPV.

The four stabilizer assemblies are located in the RPV annulus at 75, 165, 255, and 345 degrees azimuthal locations.

3.9.5.1.1.2 Shroud Support

The shroud support is designed to support the shroud and to support and locate the jet pumps. The shroud support provides an annular baffle between the reactor pressure vessel and the shroud. The jet pump discharge diffusers penetrate the shroud support to introduce the coolant to the inlet plenum below the core.

3.9.5.1.1.3 Shroud Head and Steam Separator Assembly

This component is not a core support structure or safety class component. It is discussed here to describe the coolant flow paths in the reactor pressure vessel. The shroud head and steam separator assembly is bolted to the top of the top guide 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 are attached to the top of standpipes that are welded into the shroud head. The steam separators have no moving parts. In each separator, the steam-water mixture rising through the standpipe passes vanes that impart a spin 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.

3.9.5.1.1.4 Core Plate

The core 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, in-core flux monitor guide tubes, peripheral fuel supports, and startup neutron sources. The last two items are also supported vertically by the core support plate.

The entire assembly is bolted to a support ledge on the lower portions of the shroud.

3.9.5.1.1.5 Top Guide

The top guide consists of a circular grid plate with square openings welded to the bottom of the top guide cylinder. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, fewer than four fuel assemblies. Notches are provided in the bottom of the intersections to anchor the in-core flux monitors and startup neutron sources. The top guide is bolted to the shroud. The core spray spargers are installed in the upper portion of the top guide cylinder.

3.9.5.1.1.6 Fuel Support

The fuel supports, shown in Figure 3.9-9, 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 assure proper coolant flow to the peripheral fuel assembly. Each four-lobed orificed fuel support will support four fuel assemblies and is provided with four orifice plates to assure proper coolant flow distribution to each rod-controlled fuel assembly. The four-lobed orificed fuel supports rest in the top of the control rod guide tubes, which are supported laterally by the core plate. The control rods pass through slots in the center of the four-lobed orificed fuel support. A control rod and the four adjacent fuel assemblies represent a core cell (see Subsection 4.2.2).

3.9.5.1.1.7 Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the control rod drive housings up through holes in the core plate. Each tube is designed as the guide for a control rod and as the vertical support for a four-lobed orificed fuel support piece and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the 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.

3.9.5.1.1.8 Jet Pump Assemblies

The jet pump assemblies are not core support structures but are discussed here to describe coolant flow paths in the vessel. The jet pump assemblies are located in two semicircular groups in the downcomer annulus between the core shroud and the reactor vessel wall. The design and performance of the jet pump is covered in detail in References 1 and 2. Each stainless steel jet pump consists of driving nozzles, suction inlet, throat or mixing section, and diffuser (see Figure 3.9-10). The driving nozzle, suction inlet, and throat are joined together as a removable unit, and the diffuser is permanently installed. High-pressure water from the recirculation pumps is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace consists of cantilever beams welded to a riser pipe and to pads on the reactor vessel wall.

The nozzle entry section is connected to the riser by a metal-to-metal, spherical-to-conical seal joint. Firm contact is maintained by a holddown clamp. The throat section is supported laterally by a bracket attached to the riser. There is a slipfit joint between the throat and diffuser. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

Illinois Power Company reduced the preload on the beams from 30 to 25 kips in accordance with General Electric recommendations. This increases the expected life of these beams without cracking to 19 to 40 years. Inspection frequencies were developed based on a lead plant experience and GE testing. If cracking is detected, the beams will be replaced using the best engineering solution at the time. We feel that this is a satisfactory long-term solution for BWR's. (Q&R MEB (DSER) 89)

3.9.5.1.1.9 Steam Dryers

The steam dryer assembly is not a core support structure or safety class component. It is discussed here to describe coolant flow paths in the vessel. The steam dryers remove moisture from the wet steam leaving the steam separators. The extracted moisture flows down the dryer vanes to the collecting troughs, then flows through tubes into the downcomer annulus. A skirt extends from the bottom of the dryer vane housing to the steam separator standpipe, below the water level. This skirt forms seal between the wet steam plenum and the dry steam flowing from the top of the dryers to the steam outlet nozzles.

Though the steam dryer is not a safety class component, it is required to maintain its structural integrity during normal, upset, and faulted conditions. Structural integrity of the steam dryer is assured by analysis and visual inspection. A gross visual inspection is performed each refueling outage while the dryer is being removed from the reactor and checked for any gross deformations. In addition detailed visual inspections of the steam dryer are performed in accordance with BWRVIP-139 (Reference 13).

The steam dryer and shroud head are positioned in the vessel during installation with the aid of vertical guide rods. The dryer assembly rests on steam dryer support brackets attached to the reactor vessel wall. Upward movement of the dryer assembly, which may occur under accident conditions, is restricted by steam dryer hold down brackets attached to the reactor vessel top head.

3.9.5.1.1.10 Feedwater Spargers

These components are not core support structures or safety class components. They are discussed here to describe flow paths in the vessel. The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger is fitted to each feedwater nozzle and is shaped to conform to the curve of the vessel wall.

Sparger end brackets are pinned to vessel brackets to support the spargers. Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryer before it contacts the vessel wall. The feedwater also serves to condense the steam in the region above the downcomer annulus and to subcool the water flowing to the jet pumps and recirculation pumps.

3.9.5.1.1.11 Core Spray Lines

This component is not a core support structure. It is discussed here to describe a safety class feature inside the reactor pressure vessel. The core spray lines are the means for directing flow to the core spray nozzles, which distribute coolant during accident conditions.

Two core spray lines enter the reactor vessel through the two core spray nozzles (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 top guide cylinder 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 top guide cylinder. 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 top guide 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 top guide cylinder. The correct spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from spargers (see Section 6.3).

3.9.5.1.1.12 Vessel Head Spray Nozzle

This component is not a core support structure. It is included here to describe a safety class feature in the reactor pressure vessel. 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, helps to maintain the cooldown rate.

The vessel head spray nozzle is mounted to a short length of pipe and a flange, which is bolted to a mating flange on the reactor head nozzle (see Subsection 5.4.7).

3.9.5.1.1.13 Differential Pressure and Liquid Control Line

This component is not a core support structure or safety class component. It is discussed here to describe the coolant paths in the reactor vessel. The differential pressure and liquid control lines enter the vessel through two bottom head penetrations and serve a dual function within the reactor vessel - to sense the differential pressure across the core support plate (described in Section 5.4, "Component and Subsystem Design") and to provide a path for the injection of the liquid control solution into the coolant stream. One line 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 other line terminates immediately above the core support plate and senses the pressure in the region outside the fuel assemblies.

3.9.5.1.1.14 In-Core Flux Monitor Guide Tubes

This component is not a core support structure or safety class component. It is discussed here to describe the coolant flow paths in the reactor vessel. It provides a means of positioning fixed detectors in the core as well as provide a path for calibration monitors (TIP System).

The in-core flux monitor guide tubes extend from the top of the in-core flux monitor housing (see Section 5.4) 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.

3.9.5.1.1.15 Surveillance Sample Holders

This component is not a core support structure or a safety class component. It is discussed here to describe the coolant flow paths in the reactor vessel. The surveillance sample holders are welded baskets containing impact and tensile specimen capsules. The baskets hang from the brackets that are attached to the inside wall of the reactor vessel and extend to mid-height of the active core. The radial positions are chosen to expose the specimens to the same environment and maximum neutron fluxes as experienced by the reactor vessel itself while avoiding jet pump removal interference or damage.

3.9.5.1.1.16 Low-Pressure Coolant Injection Lines

This component is not a core support structure but is discussed here to describe the coolant flow paths in the reactor vessel. Three LPCI lines penetrate the core shroud. LPCI flow is discharged through the LPCI couplings into the shroud below the top guide. A flow deflector welded to the inside of the shroud at each LPCI opening disperses the entering flow to reduce the flow forces on in-core instrumentation.

3.9.5.2 Design Loading Conditions

3.9.5.2.1 Events to be Evaluated

Examination of the spectrum of conditions for which the safety design basis must be satisfied by core support structures and engineered safety features components reveals the following four significant faulted events:

a. Recirculation Line Break: a break in a recirculation line between the reactor vessel and the suction or discharge side of the pump.

b. Steamline break accident: a break in one main steamline between the reactor vessel and the flow restrictor. The accident results in significant pressure differentials across some of the structures within the reactor.

c. Earthquake: subjects the core support structures and reactor internals to significant forces as a result of ground motion.

d. Safety relief valve discharge in combination with an SSE.

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents shows that the loads affecting the core support structures and other engineered safety feature reactor internals are less severe than those due to the above postulated events. The faulted conditions for the reactor pressure vessel internals are discussed in Subsection 3.9.1.4. Loading combination and analysis for the reactor pressure vessel internals are discussed in Subsection 3.9.3.1 and Table 3.9-1 and 3.9-2.

The core support structures are designed in accordance with Subsection NG of Section 111 of the ASME Code.

As part of the 120% Extended (Licensed) Power Uprate (LPU @ 3473 MWt), a structural integrity assessment of the key reactor internal components was performed. The thermal hydraulic analysis data, Reactor Internal Pressure Differences, and the acoustic and flow induced loads due to a postulated Recirculation line break (LOCA), including GE14 fuel, were used as input to the EPU evaluation (Ref. 11).

Since the Reactor Internal Pressure Difference increase due to EPU impacts vessel internal inspections, the NRC acceptance of EPU relies, in part, on compliance with the NRC approved BWRVIP inspection program for safety related reactor internals (Section 3.3.3 of Ref 12).

3.9.5.2.2 Pressure Differential During Rapid Depressurization

A digital computer code is used to analyze the transient conditions within the reactor vessel following the recirculation line break accident and the steam line break accident. The analytical model of the vessel consists of nine nodes, 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 3.9-11 shows the nine reactor nodes. The computer code used is the General Electric Short-Term Thermal-Hydraulic Model described in Reference 3. This model has been approved for use in ECCS conformance evaluation under 10 CFR 50, Appendix K. In order to adequately describe the blowdown pressure effect on the individual assembly components, three features are included in the model that are not applicable to the ECCS analysis and are, therefore, not described in Reference 3. These additional features are discussed in the following paragraphs:

a. The liquid level in the steam separator region and in the annulus between the dryer skirt and the pressure vessel is tracked to more accurately determine the flow and mixture quality in the steam dryer and in the steamline.

b. The flow path between the bypass region and the shroud head is more accurately modelled since the fuel assembly pressure differential is influenced by flashing in the guide tubes and bypass region for a steamline break. In the ECCS analysis, the momentum equation is solved in this flow path, but its irreversible loss coefficient is conservatively set at an arbitrary low value.

c. The enthalpies in the guide tubes and the bypass are calculated separately, since the fuel assembly WP is influenced by flashing in these regions. In the ECCS analysis, these regions are lumped.

3.9.5.2.3 Recirculation Line and Steamline Break

3.9.5.2.3.1 Accident Definition

Both a recirculation line break (the largest liquid break) and an inside steamline break (the largest steam break) are considered in determining the design basis accident for the engineered safety feature reactor internals. The recirculation line break is the same as the design basis loss-of-coolant accident described in Section 6.3. A sudden, complete circumferential break is assumed to occur in one recirculation loop. The pressure differentials on the reactor internals and core support structures are lower in all cases than for the main steamline break.

The analysis of the steamline break assumes a sudden, complete circumferential break of one main steamline between the reactor vessel and the main steamline restrictor. 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 internal structures.

The steamline break accident produces significantly higher pressure differentials across the reactor internal structures than does the recirculation line break. This results from the higher reactor depressurization rate associated with the steamline break. Therefore, the steamline break is the design basis accident for internal pressure differentials.

As part of the 120% Extended (Licensed) Power Uprate (LPU @ 3473 MWt), a structural integrity assessment of the key reactor internal components was performed. The thermal hydraulic analysis data, Reactor Internal Pressure Differences, and the acoustic and flow induced loads due to a postulated Recirculation line break (LOCA), including GE14 fuel, were used as input to the EPU evaluation (Ref. 11).

3.9.5.2.3.2 Effects of Initial Reactor Power and Core Flow

The maximum internal pressure loads can be considered to be composed of two parts: steady-state and transient pressure differentials. For a given plant, the core flow and 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 is conducted at a low-power, high-recirculation flow condition in addition to the standard safety analysis condition at high power and rated recirculation flow. The power chosen for analysis is the minimum value permitted by the recirculation system controls at rated recirculation drive flow (that is, the drive flow necessary to achieve rated core flow at rated power.)

This condition maximizes those loads which are inversely proportional to power. It must be noted that this condition, while possible, is unlikely; 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.

As part of the 120% Extended (Licensed) Power Uprate (LPU @ 3473 MWt), a structural integrity assessment of the key reactor internal components was performed. The thermal hydraulic analysis data, Reactor Internal Pressure Differences, and the acoustic and flow induced loads due to a postulated Recirculation line break (LOCA), including GE14 fuel, were used as input to the EPU evaluation (Ref. 11).

3.9.5.2.4 Seismic and Hydrodynamic Events

The seismic and hydrodynamic loads acting on the structures within the reactor vessel are based on a dynamic analysis as described in Sections 3.7, 3.8, and 3.9.2.5. Dynamic analysis is performed by coupling the mathematical model of the reactor vessel and internals with the building model to determine the accelerations, forces, and moment time histories in the reactor vessel and internals. This is done using the modal superposition method. Acceleration response spectra are also generated for subsystem analyses of selected components.

3.9.5.3 Design Bases

3.9.5.3.1 Safety Design Bases

The reactor core support structures and internals shall meet the following safety design bases:

a. Shall be arranged to provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel.

b. Deformation shall be limited to assure that the control rods and core standby cooling systems can perform their safety functions.

c. Mechanical design of applicable structures shall assure that safety design bases (a) and (b) above are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.9.5.3.2 Power Generation Design Bases

The reactor core support structures and internals shall be designed to the following power generation design bases:

a. They shall provide the proper coolant distribution during all anticipated normal operating conditions to full power operation of the core without fuel damage.

b. They shall be arranged to facilitate refueling operations.

c. They shall be designed to facilitate inspection.

3.9.5.3.3 Design Loading Categories

The basis for determining faulted loads on the reactor internals is given for seismic and hydrodynamic loads in Section 3.7, 3.8, and 3.9.2.5, and for pipe rupture loads in Subsections 3.9.5.3.2 and 3.9.5.3.4.

Core support structure and safety class internals stress limits are consistent with ASME B&PV Code Section III, "Categorization of Loading Conditions" (NA-2140) and associated stress limits contained in Addenda dated through Summer 1976. Level A, B, C and D service limits defined in the Winter 1976 Addenda which replace normal, upset, emergency and faulted condition limits are not reflected in design documentation for core support structures and other safety class reactor internals for this reactor. However, for these components, Level A, B, C and D service limits are judged to be equivalent to the normal, upset, emergency and faulted loading condition limits, and therefore, for clarity, both sets of nomenclature are retained herein.

Stress intensity and other design limits are discussed in Subsection 3.9.5.3.5. The core support structures which are fabricated as part of the reactor pressure vessel assembly are discussed in Subsection 3.9.1.4.3.

The design requirements for equipment classified as "other internals" e.g., steam dryers and shroud heads, were specified by the designer with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate. Where possible, design requirements are based on applicable industry codes and standards. If these are not available, the designer relies on accepted industry or engineering practices.

3.9.5.3.4 Response of Internals Due to Inside Steam Break Accident

The maximum pressure loads acting on the reactor internal components result from an inside steamline break, and on some components the loads are greatest with operation at the minimum power associated with the maximum 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. As part of the 120% Extended (Licensed) Power Uprate (LPU @ 3473 MWt), a structural integrity assessment of the key reactor internal components was performed. The thermal hydraulic analysis data, Reactor Internal Pressure Differences, and the acoustic and flow induced loads due to a postulated Recirculation line break (LOCA), including GE14 fuel, were used as input to the EPU evaluation (Ref. 11).

3.9.5.3.5 Stress, Deformation, and Fatigue Limits for Engineered Safety Feature Reactor Internals (Except Core Support Structure)

The stress deformation and fatigue criteria are documented in Tables 3.9-2 (a) and 3.9-2 (b).

Components inside the reactor pressure vessel such as control rods which must move during accident condition have been examined to determine if adequate clearances exist during emergency and faulted conditions. No mechanical clearance problems have been identified. The forcing functions applicable to the reactor internals are discussed in Subsection 3.9.2.5.

3.9.5.3.6 Stress and Fatigue Limits for Core Support Structures

The stress, fatigue, and other limits for the core support structures are in accordance with ASME Section III, Subsection NG. See Table 3.9-2 (a).

3.9.6 Inservice Testing of Pumps and Valves

Inservice testing of ASME Code Classes 1, 2 and 3 pumps and valves will be performed to ensure operational and functional readiness throughout their service life. The applicable edition and addenda of the Code will be as required by 10 CFR 50.55a paragraph (f), pursuant to relief granted by 10 CFR 50.55a paragraph (f) (6) (i).

A separate inservice inspection program document has been submitted, delineating the pumps and valves to be tested in accordance with the requirements of 10 CFR 50.55a paragraph (g) (5) (i).

3.9.6.1 Deleted

3.9.6.2 Deleted

3.9.6.3 Relief Requests

Requests for relief from Section XI requirements will be submitted in a format recommended by the NRC and will demonstrate either of the following:

a. Compliance with the code requirements would result in hardships or unusual difficulties without a compensating increase in the level of safety, and noncompliance will provide an acceptable level of quality and safety.

b. Proposed alternatives to the code requirements or portions thereof will provide an acceptable level of quality and safety.

3.9.6.4 Reactor Coolant System Pressure Isolation Valves

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. In order to prevent exceeding the design pressure of the low pressure systems, thus causing an intersystem LOCA, the leak-tight integrity of these valves must be ensured by periodic testing.

The following is a list of those valves which perform a pressure isolation function between the reactor coolant system (RCS) and a low pressure system:

SYSTEM	VALVE NUMBER	SERVICE	DRAWING NUMBER
LPCS	E21-F006*	LPCS Injection	M05-1073
	E21-F005*	LPCS Injection	M05-1073

SYSTEM	VALVE NUMBER	SERVICE	DRAWING NUMBER
	E21-F005*	LPCS Injection	M05-1073
HPCS	E22-F005*	HPCS Injection	M05-1074
	E22-F004*	HPCS Injection	M05-1074
RHR	E12-F041A,B,C*	LPCI Injection	M05-1075
	E12-F042A,B,C*	LPCI Injection	M05-1075
	E12-F050A,B*	Shutdown Cooling Return	M05-1075
	E12-F053A,B*	Shutdown Cooling Return	M05-1075
	E12-F009	Shutdown Cooling Suction	M05-1075
	E12-F008***	Shutdown Cooling Suction	M05-1075
	1E12-F499A/B	FWLC Supply	M05-1075
	1E12-F495A/B	FWLC Supply	M05-1075
	E12-F499A/B	FWLC Supply	M05-1075
	E12-F023**	RHR & RX Head Spray	M05-1075
	E12-F496***	FWLC Supply	M05-1075
	E12-F497***	FWLC Supply	M05-1075

SYSTEM	VALVE NUMBER	SERVICE	DRAWING NUMBER
RCIC	E51-F066***	RCIC Head Spray	M05-1079
	E51-F013**	RCIC Head Spray	M05-1079

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* These valves form a pressure isolation barrier which consists of a check valve in series with a motor-operated valve. The check valve is provided with a small (2-inch or smaller) bypass valve to allow pressure equalization for performance testing of the check valve. The sections of piping on the outboard side of the motor operated valves are provided with a safety/relief valve capable of handling the maximum postulated leakage flow through the bypass valves around the inboard check valves. Therefore, the potential for overpressurization by leakage through the bypass valves has been adequately addressed in the original design, and need not be considered further.

** These valves meet the criteria of the first note except there is no bypass valve around the check valve for pressure equalization. The check valve is provided with a lever for performance testing.

*** These valves meet the criteria of the first note except there is no bypass valve around the check valve for pressure equalization and the check valve is not provided with a lever for performance testing. 1E51-F066 serves as a check valve for both 1E12-F023 and 1E51-F013.

The additional leakage tests required by the NRC to meet these criteria are performed at an RCS pressure of greater than or equal to 1000 psig and less than or equal to 1025 psig. This is a separate test from the Appendix J, Type C tests. Therefore, no information on the methods of extrapolating from the low pressure Type C testing to this test need be supplied.

The established acceptance criteria for each valve subject to this special leak test is an equivalent leakage of less than or equal to 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm.

Although the suggested mode of shutdown prior to entering Mode 2 for performing this test appears to be reasonable, Clinton Power Station does not see the need to restrict operational flexibility by committing to a specific time for executing this test. Leak testing is normally performed during shutdown; however, it may be more desirable to perform these tests as the unit is brought down to cold shutdown or during the outage, rather than when the unit is being restarted. The determination of the exact time sequence in the refueling outage for their performance should remain unspecified to not impact outage critical path. The initial testing interval of once per refueling outage can be implemented, but it should be noted that this will not average to a yearly interval because Clinton is designed for an 24 month refueling cycle.

The pressure isolation barriers, which consist of check valves in series with motor-operated valves, have been identified by an asterisk or asterisks in the list provided in this response.

In addition, vent/drain connections are installed between the inboard isolation valve and the outboard isolation valve. These connections were originally installed in order to perform leakage testing. Prior to performing these surveillance tests, the test connections were fitted with a pressure gauge to detect any high pressure between the valves. This same procedure can be utilized in the performance of this new leakage test to ensure personnel safety.

The LPCS, RHR/LPCI and RHR/Shutdown Cooling Return and RHR Shutdown Cooling Suction lines are all monitored for RCS leakage into the system by pressure switches located in the pump discharge lines (or suction lines for shutdown cooling suction HPCS and RCIC) outside the primary containment.

These switches activate a high-pressure alarm in the main control room when the line pressure exceeds its normal high value. ECCS lines are provided with safety/relief valves which relieve any overpressure leaking from the RCS through the outboard isolation by discharging to the suppression pool. The ECCS pumps are also provided with thermal relief valves which relieve any overpressurization that leaks back through the check valves in the pump discharge line.

If any of the leakages through the RCS boundary valves were significant, the RPV water level imbalance, as well as the suppression pool level increase, resulting from the safety/relief valve discharge would provide the plant operators with additional indication that a problem has developed. It should also be noted that the piping between the inboard valve and outboard isolation valve is designed for full RCS design conditions (or greater) in all cases listed in Item 1. In the cases of the HPCS injection line, and the RCIC head spray lines, the design pressure of these lines meet or exceed the RCS design pressure on the outboard side of the outboard isolation valve. Therefore, overpressurization of these lines has been considered in the basis design. To get to a point in these systems where RCS leakage would present an overpressurization problem, the leakage of other system valves, such as the pump discharge check valves, would also have to be postulated. This additional leakage, coupled with the postulated leakages already assumed to occur through both RCS boundary pressure isolation valves, does not appear to be a plausible event.

Based on the design provisions described in the preceding paragraphs, it is felt that sufficient justification has been provided to preclude the need to perform this leakage test each time the valve is moved from its fully closed position. This post-maintenance leak testing will be performed on a particular valve only when the maintenance affects its pressure retaining capability. Otherwise, this leak test is performed by the regular interval of each refueling outage. Testing these valves each time they are moved from the fully closed position is not practical, especially in the case of the inboard check valves. Thus, exception is taken to the portion of this question concerning the requirement of leak testing when a valve is moved from its fully closed position (Q&R MEB (DSER) 87).

3.9.7 References

1. "Design and Performance of G. E. BWR Jet Pumps," General Electric Company, Atomic Power Equipment Department, APED- 5460, July 1968.

2. Moen, H.H., "Testing of Improved Jet Pumps for the BWR/6 Nuclear System," General Electric Company, Atomic Power Equipment Department, NEDO- 10602, June 1972.

3. 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.

4. "BWR Fuel Channel Mechanical Design and Deflection," NEDE-21354-P, September 1976.

5. "BWR Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings," NEDE-21175-3-P, July 1982.

6. "Preservice Inspection and Testing of Snubbers," Robert L. Tedesco (NRC) letter to G. E. Wuller (IP), dated February 13, 1981.

7. CPS Seismic Qualification Package SQ-CL711, Revision 1, “Dynamic Qualification of Standby Liquid Control Valves”, dated April 10, 1986.

8. CPS Seismic Qualification Package SQ-CL709, Revision 4, “Dynamic Qualification of Limitorque Valve Actuators”, dated March 3, 1986.

9. NEDE-10313, PDA-Pipe Dynamic Analysis Program for Pipe Rupture Movement (Proprietary Filing).

10. NEDE-21821-A, February 1980, Boiling Water Reactor Feedwater Nozzle/Sparger Final Report (Proprietary Filing).

11. “Safety Analysis Report for Clinton Power Station Extended Power Uprate,” NEDC-32989P, June 2001.

12. Letter dated April 5, 2002. “Clinton, Unit 1, Amendment No. 149, Power Uprate (Non-Proprietary Safety Evaluation).” ADAMS Accession No. ML021650543.

13. BWRVIP-139, “BWR Vessel and Internals Project, Steam Dryer Inspection and Flaw Evaulation Guidelines.”

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TABLE 3.9-1
PLANT EVENTS

NORMAL, UPSET, AND TESTING CONDITIONS		NO. OF CYCLES
1.	Bolt Up (1)	123
2.	Design Hydrostatic Test	40
	a. Leak checks at 400 psig prior to power operation, 3 cycles/startup	
3.	Startup (100°F/hr Heatup Rate) (2)	120
4.	Daily Reduction to 75% Power(1)	10,000
5.	Weekly Reduction 50% Power(1)	2,000
6.	Control Rod Pattern Change(1)	400
7.	Loss of Feedwater Heaters (80 Cycles Total):	80
8.	OBE at Rated Operating Conditions	10/50(4)
9.	Scram:	
	a. Turbine Generator Trip, Feedwater On, Isolation Valves Stay Open	40
	b. Other Scrams	140
	c. Loss of Feedwater Pumps, Isolation Valves Closed	10
	d. Turbine Bypass, Single Safety or Relief Valve Blowdown	8
10.	Reduction to 0% Power, Hot Standby, Shutdown (100° F/hr Cooldown Rate)	111
11.	Unbolt(1)	123
12.	Single Loop Operation (Recirculation) - each loop	50

TABLE 3.9-1(Cont'd)

EMERGENCY CONDITIONS		NO. OF CYCLES
1.	Scram:	
	a. Reactor Overpressure with Delayed Scram, Feedwater Stays On, Isolation Valves Stay Open	1(3)
	b. Automatic Depressurization System Actuation	1(3)
2.	Improper Start of Cold Recirculation Loop	1(3)
3.	Sudden Start of Pump in Cold Recirculation Loop	1(3)
4.	Improper Pump Startup from Hot Standby with Reactor Drain Shut Off Followed by Pump Restart	1(3)
FAULTED CONDITION		
1.	Pipe Rupture and Blowdown (5)	1(3)
2.	Safe Shutdown Earthquake at Rated Operating Conditions	1(3)

(1) Applies to reactor pressure vessel only.

(2) Bulk average vessel coolant temperature change in any 1-hour period.

(3) The annual encounter probability of the one cycle event is $<10^{-2}$ for emergency and $<10^{-4}$ for faulted type events.

(4) Fifty peak OBE cycles for NSSS piping and ten-peak OBE cycles for other NSSS equipment and components. For BOP Design 50 Maximum Load Cycles are Used for Design as Specified in 3.7.3.

(5) The ECCS injection is a part of the scenario related to the pipe rupture and blowdown event. The thermal shock effects are considered as a consequence of the pipe rupture.

TABLE 3.9-1(Cont'd)

In addition to the above temperature/pressure/flow transients the following dynamic load transients have been considered to the design and/or fatigue evaluation:

Dynamic/Transient Load	Category	Cycles/Actuations/Events
1. Operating Basis Earthquake (OBE)	Upset	10 cycles
2. Safe Shutdown Earthquake (SSE) (6)	Faulted	1 cycle
3. Turbine Stop Valve Closure (TSV) (7)	Upset	660 cycles
4. Safety Relief Valve Actuation (Acoustic wave) (7)(8)	Upset	5460 cycles
5. Safety Relief Valve Actuation	Upset	(a) Piping Analysis 271 Events (i) All Valve 271 Actuations (ii Single Valve 1512 Actuations (b) RPV & Internals Analysis 12,600 cycles
6. Loss of Coolant Accident (LOCA):		
Small break LOCA	Emergency/faulted	1 event
Intermediate break LOCA	Faulted	1 event
Large break LOCA	Faulted	1 event

---

(6) One Stress reversal cycle of maximum seismic amplitude.

(7) Applicable to main steam piping system only. (1820 actuations with three acoustic cycles each.)

(8) 5460 cycles based on 1820 actuations with three acoustic cycles each.

TABLE 3.9-1(a)
THERMAL CYCLES FOR RPV NOZZLES AND PIPING ANALYSIS

Event No.	Plant Description	Remarks
1	Normal, Upset Test	Bolt Up
2	“	Design Hydro Test
3	“	Startup
4	“	Turbine Roll Increase to Rated Power
5	“	Daily Reduction 75%
6	“	Weekly Reduction 50%
7	“	Rod Pattern Change
8	“	Turbine Trip with 100% Steam By-Pass
9	“	Partial Feedwater Heater By-Pass
10	“	Turbine Generator Trip Feedwater on
11	“	Other Scrams
12	“	Rated Power Normal Operation
13	“	Reduction to 0% Power
14	“	Hot Stand-By
15	“	Shutdown
16	“	Shutdown Vessel Flooding
17	“	Shutdown
18*	“	Unbolt
19	“	Refueling
20	“	Composite Loss of Feedwater Pumps, Loss of Auxiliary Power, and Turbine Generator Trip without By-Pass
21	“	Turbine By-Pass, Single Relief or Safety Valve Blowdown
22	Emergency	Reactor Overpressure with Delayed Scram, Feedwater Stays On, Isolation Valves Stays Open
23	“	Automatic Blowdown
24	“	Improper Start of Cold Recirculation Loop
25	“	Sudden Start of Pump in Cold Recirculation Loop
26	“	Hot Standby Drain Shut-Off Pump Restart
27	Faulted	Pipe Rupture and Blowdown
28	Normal	Single Loop Operation

*Applies to Reactor Vessel only.

TABLE 3.9-1(a)
THERMAL CYCLES FOR RPV NOZZLES AND PIPING ANALYSIS (Continued)

Recirculation Inlet	Recirculation Outlet	Steam Outlet	Feedwater
Event No. & Cycles	Event No. & Cycles	Event No. & Cycles	Event No. & Cycles
1 (123)	1 (123)	1 (123)	1 (123)
2 (40)	2 (40)	2 (40)	2 (40)
3 (120)	3 (120)	3 (120)	3 (120)
4 (120)	4 (120)	4 (120)	4 (120)
5 (10,000)	5 (10,000)	5 (10,000)	5 (10,000)
6 (2,000)	6 (2,000)	6 (2,000)	6 (2,000)
7 (400)	7 (400)	7 (400)	7 (400)
8 (10)	8 (10)	8 (10)	8 (10)
9 (70)	9 (70)	9 (70)	9 (70)
10 (40)	10 (40)	10 (40)	10 (40)
11 (140)	11 (140)	11 (140)	11 (140)
12 (-)	12 (-)	12 (-)	12 (-)
13 (111)	13 (111)	13 (111)	13 (111)
14 (111)	14 (111)	14 (111)	14 (111)
15 (111)	15 (111)	15 (111)	15 (111)
16 (111)	16 (111)	16 (111)	16 (111)
17 (111)	17 (111)	17 (111)	17 (111)
18 (123)	18 (123)	18 (123)	18 (123)
19 (-)	19 (-)	19 (-)	19 (-)
20 (10)	20 (10)	20 (10)	20 (10)
21 (8)	21 (8)	21 (8)	21 (8)
22 (1)	22 (1)	22 (1)	22 (1)
23 (1)	23 (1)	23 (1)	23 (1)
24 (1)	24 (1)	24 (1)	24 (1)
24 (1) Note 1			
25 (1)	25 (1)	25 (1)	25 (1)
26 (1)	26 (1)	26 (1)	26 (1)
27 (1)	27 (1)	27 (1)	27 (1)
28 (50)	28 (50)		

TABLE 3.9-1(a)
THERMAL CYCLES FOR RPV NOZZLES AND PIPING ANALYSIS (Continued)

Head Cooling Spray Event No. & Cycles	Core Spray (Low Press) Event No. & Cycles	Core Spray (High Press) Event No. & Cycles	RHR/LPCI Mode Event No. & Cycles
1 (123)	1 (123)	1 (123)	1 (123)
2 (40)	2 (40)	2 (40)	2 (40)
3 (120)	3 (120)	3 (120)	3 (120)
4 (120)	4 (120)	4 (120)	4 (120)
5 (10,000)	5 (10,000)	5 (10,000)	5 (10,000)
6 (2,000)	6 (2,000)	6 (2,000)	6 (2,000)
7 (400)	7 (400)	7 (400)	7 (400)
8 (10)	8 (10)	8 (10)	8 (10)
9 (70)	9 (70)	9 (70)	9 (70)
10 (40)	10 (40)	10 (40)	10 (40)
11 (140)	11 (140)	11 (140)	11 (140)
12 (-)	12 (-)	12 (-)	12 (-)
13 (111)	13 (111)	13 (111)	13 (111)
14 (111)	14 (111)	14 (111)	14 (111)
15 (111)	15 (111)	15 (111)	15 (111)
16 (111)	16 (111)	16 (111)	16 (111)
17 (111)	17 (111)	17 (111)	17 (111)
18 (123)	18 (123)	18 (123)	18 (123)
19 (-)	19 (-)	19 (-)	19 (-)
20 (41)	20 (10)	20 (10)	20 (10)
21 (8)	21 (8)	21 (8)	21 (8)
22 (1)	22 (1)	22 (1)	22 (1)
23 (1)	23 (1)	23 (1)	23 (1)
24 (1)	24 (1)	24 (1)	24 (1)
	24[Note 1]	24[Note 1]	24[Note 1]
25 (1)	25 (1)	25 (1)	25 (1)
26 (1)	26 (1)	26 (1)	26 (1)
27 (1)	27 (1)	27 (1)	27 (1)
--- (40)[Note 2]	--- (10)[Note 4]	--- (10)[Note 3]	--- (10)[Note 5]

TABLE 3.9-1(a)
THERMAL CYCLES FOR RPV NOZZLES AND PIPING ANALYSIS (Continued)

Control Rod Drive Event No. & Cycles	Control Rod Drive Event No. & Cycles	CRD HYDR Return Event No. & Cycles	Drain (with Flow) Event No. & Cycles
1 (123)	25(1)	1 (123)	1 (123)
2 (40)	26(1)	2 (40)	2 (40)
3 (120)	27(1)	3 (120)	3 (120)
4 (120)		4 (120)	4 (120)
5 (10,000)		5 (10,000)	5 (10,000)
6 (2,000)		6 (2,000)	6 (2,000)
7 (400)		7 (400)	7 (400)
8 (10)		8 (10)	8 (10)
9 (70)		9 (70)	9 (70)
10 (40)		10 (40)	10 (40)
11 (140)		11 (140)	11 (140)
12 (-)		12 (-)	12 (-)
(50)		13 (111)	13 (111)
(-)		14 (111)	14 (111)
(10)		15 (111)	15 (111)
13 (111)		16 (111)	16 (111)
14 (111)		17 (111)	17 (111)
15 (111)		18 (123)	18 (123)
16 (111)		19 (-)	19 (-)
17 (111)		20 (10)	20 (10)
18 (123)		21 (8)	21 (8)
19 (-)		22 (1)	22 (1)
20 (10)		23 (1)	23 (1)
21 (8)		24 (1)	24 (1)
22 (1)		25 (1)	25 (1)
23 (1)		26 (1)	26 (1)
24 (1)		27 (1)	27 (1)

Note 1: This resuls from reverse flow through the recirculation outlet nozzle on improper start of cold shutdown recirculation loop.

Note 2: Can happen any time during normal operation.

Note 3: Can happen any time during rated power normal operation.

Note 4: Occurs during start up or shutdown when pressure is below 350 psig, fluid temperature in vessel is below 470 °F. Following start of this event, fluid temperature in vessel will hold at 470 °F during start up or will continue to drop at 100 °F/Hr during shutdown.

Note 5: Occurs during start up or shutdown when pressure is below 350 psig, fluid temperature in vessel is below 436 °F. Following start of this event, fluid temperature in vessel will hold at 436 °F during start up or will continue to drop at 100 °F/Hr during shutdown.

TABLE 3.9-1(b)
REACTOR VESSEL CYCLIC AND TRANSIENT LIMITS

CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
120 Heatup or Cooldown Cycles	70°F to 560°F to 70°F
80 Step Change Cycles	Loss of Feedwater Heaters
180 Reactor Trip Cycles	100% to 0% of Rated Thermal Power
40 Hydrostatic Pressure or Leak Tests	Pressurized to ≥930 psig and ≤1250 psig

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TABLE 3.9-2

DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 1, 2, AND 3 COMPONENTS

LOAD(1) CASE	N	SRV $_X$(4)	SRV ADS	OBE	SSE	SBA/IBA(3)	DBA	ASME CODE SERVICE LIMITS
1	X	X						B
2	X	X		X				C
3	X	X			X			D(2)
4	X		X			X(SBA only)		C(2)
5	X		X	X		X		D(2)
6	X		X		X	X		D(2)
7	X		X		X		X	D(2)
8	X							A
9	X			X				B

NOTES:

(1) See Load Definitions Legend for definition of terms.

(2) All ASME Code Class 1, 2, and 3 piping systems which are required to function for safe shutdown under the postulated events shall meet the requirements of NRC's "Interim Technical Position Functional Capability of Passive Piping Components."

(3) SBA or IBA, whichever is greater, except Case 4.

(4) SRV_{ALL} or SRV_1 - whichever is controlling will be used.

TABLE 3.9-2 (Cont'd)

LOAD DEFINITIONS LEGEND

N	-	Normal load consists of pressure, dead weight and thermal loads.
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_{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 valved associated with automatic depressurization system which actuate within milliseconds of each other during the postulated small or intermediate size pipe rupture.
DBA	-	Design basis accident is the sudden break of the main steam or recirculation lines (largest postulated breaks). DBA-related loads include main vent clearing and pool swell, chugging, condensation oscillation, and annulus pressurization.
SBA	-	Small-break accident.
IBA	-	Intermediate-break accident.

TABLE 3.9-2 (Cont'd)

INDEX

3.9-2(a)	Reactor Pressure Vessel and Shroud Support Assembly
3.9-2(b)	Reactor Vessel Internals and Associated Equipment
3.9-2(c)	Reactor Water Cleanup Heat Exchangers
3.9-2(d)	Class I Main Steam Piping and Pipe-Mounted Equipment
3.9-2(e)	Class I Recirculation Loop Piping and Pipe-Mounted Equipment
3.9-2(f)	Recirculation Flow Control Valve
3.9-2(g)	Safety/Relief Valves (Main Steam)
3.9-2(h)	Main Steam Isolation Valve
3.9-2(i)	Recirculation Pump
3.9-2(j)	Reactor Recirculation System Gate Valves
3.9-2(k)	Class III Safety/Relief Valve Discharge Piping
3.9-2(l)	Standby Liquid Control Pump
3.9-2(m)	Standby Liquid Control Tank
3.9-2(n)	ECCS Pumps
3.9-2(o)	RHR Heat Exchanger
3.9-2(p)	RWCU Pump
3.9-2(q)	RCIC Turbine
3.9-2(r)	RCIC Pump
3.9-2(s)	Reactor Refueling and Servicing Equipment
3.9-2(t)	CRD Housing Support
3.9-2(u)	Control Rod Drive
3.9-2(v)	Control Rod Drive Housing
3.9-2(w)	Jet Pumps
3.9-2(x)	LPCI Coupling
3.9-2(y)	Control Rod Guide Tube
3.9-2(z)	Incore Housing
3.9-2(aa)	HPCS Valves

Note: The allowable stresses in these tables are the original design basis values. The use of revised allowable stress values for ASME Section III, Class 2 and 3 piping components within the 2001 Edition through 2003 Addenda has been reconciled and is acceptable for design evaluations, modifications, repairs, and replacements.

TABLE 3.9-2(a)
REACTOR PRESSURE VESSEL AND SHROUD SUPPORT ASSEMBLY
(i) VESSEL SUPPORT SKIRT

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA		LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
Material: SA 533 GRB CLI					
A.	Normal & upset condition:				
	$P_m \leq S_m$	Normal	Primary membrane	26,700	26,550
	Sm = 26,700 @ 575° F	Pressure			
	$P_L + P_b \leq 1.5$ Sm	OBE	Primary membrane plus bending	40,100	39,900
	2.55m = 40,100 @ 575° F	SRV			
B.	Emergency condition:				
	$r_m \leq$ Sy	Normal	Primary membrane	42,800	31,010
	Sy = 42,800 @ 575° F	Pressure			
	$P_L + P_b \leq 1.5$ Sy	Chugging	Primary membrane plus bending	64,300	53,300
	1.5 Sy = 64,300 @ 575° F	SRV			
C.	Faulted condition:				
	$P_m \leq$ Sy	Normal	Primary membrane	56,000	31,010
	Sy = 42,800 @ 575° F	Pressure			
	$P_L + P_b \leq 1.5$ Sy	Scram	Primary membrane plus bending	84,000	53,300
	1.5 Sy = 64,300 @ 575° F	SSE			
D.	Maximum cumulative usage factor: 0.88 at vessel-skirt junction weld.				

TABLE 3.9-2 (a) (Cont'd)

(ii) SHROUD SUPPORT

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA		LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
Material: SB 168					
A.	Normal & upset condition:				
	$P_m \leq 0.9$ Sm	Normal	Primary membrane	20,970	15,500
	Sm = 23,300 @ 575° F	Pressure			
	$P_L + P_b \leq 1.5$ Pm	OBE	Primary membrane plus bending	31,450	16,800
	1.5 Pm = 31,450 @ 575° F	SRV			
B.	Emergency condition:				
	$P_m \leq 0.0$ Sm	Normal	Primary membrane	20,970	15,500
	Sm = 23,300 @ 575° F	Pressure			
	$P_L + P_b \leq 1.5$ Pm	SRV	Primary membrane plus bending	31,450	16,800
	1.5 Pm = 31,450 @ 575° F	Chugging			
C.	Faulted condition:				
	$P_m \leq 0.7 \times 0.9$ Su	Normal	Primary membrane	46,600	30,400
	Su = 74,300 @ 575° F	Pressure			
		SRV			
	$P_L + P_b \leq 1.5$ Pm	SSE	Primary membrane plus bending	69,900	47,100
	1.5 Pm = 69,900 @ 575° F	Chugging			

D. Maximum cumulative usage factor: 0.41 at shroud support plate.

TABLE 3.9-2 (a) (Cont'd)

(iii) RPV FEEDWATER NOZZLE

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA		LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
Material: SA-508 CLI					
A.	Normal & upset condition:				
	$P_m \leq Sm$	Normal	Primary membrane	17,700	16,220
	Sm = 17,700 @ 575° F	Pressure			
	$P_L + P_b \leq 1.5$ Sm	OBE	Primary membrane plus bending	26,550	21,300
	1.5 Sm = 26,550 @ 575° F	SRV			
B.	Emergency condition:				
	$P_m \leq Sy$	Normal	Primary membrane	25,900	21,420
	Sy = 25,900 @ 594° F	Pressure			
	$P_L + P_b \leq 1.5$ Sy	Chugging	Primary membrane plus bending	38,850	27,000
	1.5 Sy = 38,850 @ 594° F	SRV			
C.	Faulted condition:				
	$P_m \leq Sy$	Normal	Primary membrane	25,900	21,420
	Sy = 25,900 @ 594° F	Pressure			
	$P_L + P_b \leq 1.5$ Sy	Chugging	Primary membrane plus bending	38,850	27,000
	1.5 Sy = 38,850 @ 594° F	SRV			
		SSE			
D.	Maximum cumulative usage factor:	< 1.0 at Nozzle Blend Radius.			

TABLE 3.9-2 (a) (Cont'd)

(iv) CRD PENETRATION

ASME B&PV CODE SEC. III SUBSECTION NB PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS(psi)	MAXIMUM CALCULATED STRESS (psi)
Material: INCONEL SB167				
A. Normal & upset condition:				
$P_m \leq Sm$	Normal	Primary membrane	20,000	8,490
Sm = 20,000 @ 550° F	Pressure			
$P_L + P_b \leq Sy$	OBE	Primary membrane plus bending	30,000	15,200
Sy = 24,500 @ 528° F	SRV			
B. Emergency condition:				
$P_m \leq Sy$	Normal	Primary membrane	24,100	10,800
Sy = 24,500 @ 528° F	Pressure			
$P_L + P_b \leq 1.5$ Sy	Chugging	Primary membrane plus bending	36,150	20,100
1.5 Sy = 36,700 @ 528° F	SRV OBE			
C. Faulted condition:				
$P_M \leq Sy$	Normal	Primary membrane	48,000	10,800
Sy = 24,500 @ 528° F	Pressure			
$P_L + P_b \leq 1.5$ Sy	LOCA	Primary membrane plus bending	72,000	20,100
1.5 Sy = 36,700 @ 528° F	Scram			
	SSE			

D. Maximum cumulative usage factor: 0.41 at lower weld between tube and vessel head.

TABLE 3.9-2(b)
REACTOR INTERNALS AND ASSOCIATED EQUIPMENT
(i) GRID - BEAM WITH HIGHEST STRESS

ASME B&PV CODE SEC. III PRIMARY STRESS LIMIT CRITERIA		LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
Material: 304 L SS					
A.	Normal & upset condition:				
	$P_m \leq 1.0$ Sm	Normal	Primary membrane	14,300	8,607
	S_m = 14,300 @ 550° F	Pressure			
	$P_L + P_b \leq 1.5$ Sm	OBE	Primary membrane plus bending	21,450	21,240
	1.5 Sm = 21,450 @ 550° F	SRV			
B.	Emergency condition:				
	$P_m \leq 1.5$ Sm	Normal	Primary membrane	21,450	8,607
	S_m = 14,300 @ 550° F	Pressure			
	$P_L + P_b \leq 1.5 \times 1.5$ Sm	Chugging	Primary membrane plus bending	32,175	21,240
	2.25 Sm = 32,175 @ 550° F	SRV			
C.	Faulted condition:				
	$P_m \leq 2.4$ Sm	Normal	Primary membrane	34,320	17,216
	Sm = 14,300 @ 550° F	Pressure			
	$P_L + P_b \leq 1.5 \times 2.4$ Sm	LOCA	Primary membrane plus bending	51,480	49,890
	3.6 Sm = 51,480 @ 550° F	SSE			
		SRV			

D. Maximum cumulative usage factor: 0.52 at grid cylinder junction.

TABLE 3.9-2 (b) (Cont'd)

(ii) CORE PLATE (LIGAMENT IN TOP PLATE)

ASME B&PV CODE SEC. III PRIMARY STRESS LIMIT CRITERIA		LOAD CASE NUMBER*	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
Material: 304 L					
A.	Normal & upset condition:	Normal Condition Loads:			
	$P_m \le Sm$	1. Normal loads	Primary membrane	14,300	7,079
	S_m = 14,300 @ 550° F	2. Upset Pressure			
	$P_L + P_b \le 1.5$ Sm	3. OBE	Primary membrane plus bending	21,450	14,445
	Sm = 14,300 psi @ 550° F	4. SRV			
B.	Emergency condition:	Emergency Condition Loads:			
	$P_m \le 1.5$ Sm	1. Normal loads	Primary membrane	21,450	7,079
	Sm = 14,300 psi @ 550° F	2. Upset Pressure			
	$P_L + P_b \le 2.25$ Sm	3. Chugging	Primary membrane plus bending	32,175	14,445
	Sm = 14,300 psi @ 550° F**	4. SRV_{ADS}			
C.	Faulted condition:	Faulted Condition Loads:			
	$P_m \le 2.4$ Sm	1. Normal loads	Primary membrane	34,320	26,620
	Sm = 14,300 psi @ 550° F	2. Accident Pressure			
	$P_L + P_b \le 3.6$ Sm	3. Chugging	Primary membrane plus bending	51,480	36,906
	Sm = 14,300 psi @ 550° F**	4. SRV_{ADS}			
		5. SSE			

TABLE 3.9-2 (b) (Cont'd)

(ii) CORE PLATE (LIGAMENT IN TOP PLATE)

ASME B&PV CODE SEC. III PRIMARY STRESS LIMIT CRITERIA	LOAD CASE NUMBER*	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
D. Maximum cumulative usage factor: 0.84 at stiffener to skirt weld.				

* Load cases are defined in Table 3.9-2.

**Value of Sm or Sy is shown depending upon the controlling criteria (e.g., 1.8 Sm or 1.5 Sy for B).

TABLE 3.9-2 (b) (Cont'd)

(iii) VENT & HEAD SPRAY NOZZLE

ASME B&PV CODE SEC. III PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
Material: SA333 Carbon Steel				
A. Normal & upset condition:				
	Normal			
	Pressure			
$S_{limit} \leq 3$ Sm	OBE	Primary & secondary membrane plus bending	53,100	47,400
	SRV			
Sm = 17,700 @ 550° F				
B. Emergency condition:				
	Normal			
	Pressure			
$P_L + P_b \leq 2.25$ Sm	Chugging	Primary membrane plus bending	39,800	25,500
	SRV			
2.25 Sm = 39,800 @ 550° F				
C. Faulted condition:				
	Normal			
	Pressure			
$P_L + P_b \leq 3.6$ Sm	SSE	Primary membrane plus bending	63,700	52,300
3.6 Sm = 63,700 @ 550° F	LOCA			
	SRV			

TABLE 3.9-2 (b) (Cont'd)

(iv) ORIFICED FUEL SUPPORT

ASME B&PV CODE SEC. III, SUBSECTION NG & APPENDIX F	MODE OF LOADING	LOADING	ALLOWABLE LOAD (LB)	MAXIMUM CALCULATED LOAD (LB)
A. Normal, upset & emergency conditions	Horizontal load	Pressure	3,116	1,783
		Component weight		
Limit = .44 L_u per		OBE &SRV		
NG-3228.4 $(1)^u(2)$				
	Vertical load	Pressure	16,748	15,648
		Component weight		
		OBE & SRV		
		SCRAM		
B. Faulted condition:	Horizontal load	Pressure	5,665	2,111
		Component weight		
Limit = .80 P_t per f-1380 (2)				
		SSE & LOCA (AP+ jet reaction)		
	Vertical load	Pressure	30,451	20,981
		Component weight		
		SSE & SRV_{max}		
		SCRAM		

NOTES: (1) Normal, upset and emergency loads are combined together and compared to the upset allowable limit.

(2) These criteria depend upon test loads L_u and P_t which account for differences which may exist in actual parts and tested parts to include dimensional thickness, yield strength and casting quality.

TABLE 3.9-2 (b) (Cont'd)

(v) CORE SPRAY LINE AND SPARGER

ASME B&PV CODE SEC. III PRIMARY STRESS LIMIT CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	MAXIMUM CALCULATED STRESS (psi)
Materials: 304SS & 316L				
A. Normal & upset condition:				
^{S}limit ≤ 3 Sm Sm = 13,950 @ 550° F (304SS)		Primary membrane plus bending plus secondary membrane	41,850	31,636
B. Emergency condition:				
$P_L + P_b \leq 2.25$ Sm		Primary membrane plus bending	31,380	7,204
Sm = 13,950 @ 550° F (304SS)				
C. Faulted condition:				
$P_L + P_b \leq 3.6$ Sm		Primary membrane plus bending	40,680	38,651
Sm = 16,950 @ 550° F (316L)				

TABLE 3.9-2 (b) (Cont'd)

(vi) FUEL ASSEMBLY (INCLUDING CHANNEL)[5]

Acceptance Criteria	Loading	Primary Load Type	Calculated Peak Acceleration	Evaluation Basis Acceleration
Acceleration Envelope	Horizontal Direction: 1. Peak Pressure 2. Safe Shutdown Earthquake 3. Annulus Pressurization	Horizontal Acceleration Profile	0.9G	(1)
	Vertical Direction: 1. Peak Pressure 2. Safe Shutdown Earthquake 3. Safety Relief Valve 4. Scram	Vertical Accelerations	5.5G[4]	(1)

NOTES:

(1) Evaluation Basis Accelerations and Evaluations are contained in NEDE-21175-3-P.

(2) The calculated maximum fuel assembly gap opening for the most limiting load combination is 0.15[4] inch.

(3) The fatigue analysis indicates that the fuel assembly has adequate fatigue capability to withstand the loadings resulting from multiple SRV actuations and OBE+SRV event.

(4) These values are determined using the methodology contained in NEDE-21175-3-P.

(5) Extended Power Uprate (EPU) has insignificant effect on the seismic/dynamic response. The EPU differential pressure loads have been determined to be within the design limits of the fuel assembly.

TABLE 3.9-2(c)
TABLE DELETED

TABLE 3.9-2(d)
ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT

ACCEPTANCE CRITERIA	HIGHEST CALCULATED (1) STRESS OR USAGE FACTOR	ALLOWABLE LIMITS	RATIO CALCULATED/ ALLOWABLE	LOADING	IDENTIFICATION OF LOCATION OF HIGHEST STRESS POINTS
ASME B&PV Code Section III, NB-3600					
Primary Stress Eq 9 ≤ 1.5 Sm design condition	13093 psi	26550 psi	0.39	1. Normal loads 2. Pressure 3. Transient	4th elbow near Main Steam Line D guide lug
Primary Stress Eq 9 ≤ 1.8 Sm Service Level B	18092 psi	31860 psi	0.57	1. Normal loads 2. Pressure 3. OBE 4. SRV 5. Transient	Main Steam Line C guide lug
Primary Stress Eq 9 ≤ 2.25 Sm Service Level C	18050 psi	39825 psi	0.45	1. Normal loads 2. Pressure 3. LOCA 4. SRV 5. Transient	Main Steam Line C guide lug
Primary Stress Eq 9 ≤ 3.0 Sm Service Level D	40063 psi	54600 psi	0.73	1. Normal loads 2. Pressure 3. SSE 4. LOCA 5. Transient	SRV sweepolet Main Steam Line C
Secondary Stresses Eq 12 ≤ 3.0 Sm	44543 psi	53100 psi	0.84		Main Steam Line D 1st elbow
Primary plus secondary stress except thermal expansion Eq 13 ≤ 3.0 Sm	30664 psi	54600 psi	0.56		Main Steam Line A RCIC Tee
Fatique Usage Factor U ≤ 1.0	0.09	1.0	0.09		SRV Sweepolet Main Steam Line C

Note: Piping within the scope of this table has been evaluated for the effects of extended power uprate (EPU) and satisfy the applicable Code requirements. A summary of the Reactor Coolant Piping EPU evaluation is summarized in PUSAR NEDC-32989P.

TABLE 3.9-2 (d) Continued

ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT

COMPONENT/ LOAD TYPE	CALCULATED (1) LOAD (lbf)	ALLOWABLE LOAD (lbf)	RATIO CALCULATED/ ALLOWABLE	LOADING	IDENTIFICATION OF EQUIPMENT WITH HIGHEST LOADS RATIO
Snubbers					
Normal and upset condition (Level A&B)	11766	30000	0.392	1. Pressure 2. Weight 3. OBE 4 Turbine Stop Valve Closure	MS Line C, S104
Emergency condition (Level C)	24710	93100	0.265	1. Pressure 2. Weight 3. Chugging 4. SRV	MS Line C, S106
Faulted condition (Level D)	38008	45000	0.845	1. Pressure 2. Weight 3. Annulus pressurization 4. SSE	MS Line C, S104

TABLE 3.9-2 (d) Continued

ASME CODE CLASS 1 MAIN STEAM PIPING AND PIPE MOUNTED EQUIPMENT

COMPONENT/ LOAD TYPE	HIGHEST CALCULATED (1) LOAD	ALLOWABLE LOAD	RATIO CALCULATED/ ALLOWABLE	LOADING	IDENTIFICATION OF EQUIPMENT WITH HIGHEST LOADS
Service Level D SR Valve-horizontal acceleration	8.8g	9.0g	0.980	1. Pressure 2. Weight 3. Annulus Pressurization 4. SSE	MS Line D SR valve operator
Service Level D SR Valve-vertical acceleration	4.3g	6.0g	0.720	1. Pressure 2. Weight 3. Annulus Pressurization 4. SSE	MS Line A SR valve operator
Service Level D MSIV Bonnet/moment	964148 in-lb	1511959 in-lb	0.656	1. Pressure 2. Weight 3. SSE 4. Thermal 5. Annulus Pressurization	MS Line C MSIV
Service Level D SRV Flange moment	916403 in-lb	1474139 in-lb	0.620		MS Line C SRV inlet
Service Level D SRV Flange moment	185499 in-lb	599289 in-lb	0.310		MS Line C SRV outlet

Note: (1) Appropriate loading combination of Table 3.9-2 were considered and the calculated stresses are reported for the governing loads.

TABLE 3.9-2(e)
ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT

ACCEPTANCE CRITERIA	HIGHEST CALCULATED (1) STRESS OR USAGE FACTOR	ALLOWABLE LIMITS	RATIO CALCULATED/ ALLOWABLE	LOADING	IDENTIFICATION OF LOCATION OF HIGHEST STRESS POINTS
ASME B&PV Code Section III, NB-3600					
Primary Stress Eq 9 ≤ 1.5 Sm design condition	13,139 psi	25,013 psi	0.53	1. Normal loads	Loop B at Hangers H305 and H306 Location
Primary Stress Eq 9 ≤ 1.8 Sm Service Level B	21,958 ps	28,596 psi	0.77	1. Normal loads 2. OBE 3. SRV	RHR Tee Loop B
Primary Stress Eq 9 ≤ 2.25 Sm Service Level C	21,949 psi	34,315 psi	0.64	1. Normal loads 2. Pressure 3. SRV	RHR Tee Loop B
Primary Stress Eq 9 ≤ 3.0 Sm Service Level D	37,800 psi	38,128 psi	0.99	1. Normal loads 2. Pressure 3. SSE 4. Annulus pressurization	RHR Tee Loop B
Secondary Stresses Eq 12 ≤ 3.0 Sm	36,445 psi	50,025 psi	0.73		RHR Tee Loop B
Primary plus secondary stress except thermal expansion Eq 13 ≤ 3.0 Sm	37,598 psi	50,025 psi	0.75		Loop B Elbow end at Pump Inlet
Fatique usage Factor U ≤ 1.0	0.30	1.0	0.30		RHR Tee Loop B

TABLE 3.9-2(e) Continued

ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT

COMPONENT/ LOAD TYPE	CALCULATED (1) LOAD (lbf)	ALLOWABLE LOAD (lbf)	RATIO CALCULATED/ ALLOWABLE	LOADING	IDENTIFICATION OF EQUIPMENT WITH HIGHEST LOADS RATIO
Snubbers					
Normal and upset condition (Level A&B)	39,609	70,000	0.566	1. Normal loads 2. SRV 3. OBE	Loop B, S371
Emergency conditions (Level C)	39,742	93,100	0.427	1. Normal Loads 2. SRV 3. Chugging	Loop B, S374
Faulted conditions (Level D)	104,631	105,000	0.996	1. Normal loads 2. Annulus pressurization 3. SSE	Loop B, S371

TABLE 3.9-2(e) Continued

ASME CODE CLASS 1 RECIRCULATION PIPING AND PIPE MOUNTED EQUIPMENT

COMPONENT/ LOAD TYPE	CALCULATED (1) LOAD (lbf)	ALLOWABLE LOAD (lbf)	RATIO CALCULATED/ ALLOWABLE	LOADING	IDENTIFICATION OF EQUIPMENT WITH HIGHEST LOADS RATIO
Struts:					
Normal and upset condition (Level A&B)	38,819	50,000	0.776	1. Normal loads 2. SRV 3. OBE	B302 At recirc pump
Emergency condition (Level C)	35,672	66,500	0.536	1. Normal loads 2. SRV 3. Chugging	B302 At recirc pump
Faulted condition (Level D)	89,107	94,500	0.943	1. Normal loads 2. Annulus pressurization 3. SSE	B302 At recirc pump

Note: (1) Appropriate loading combinations of Table 3.9-2 were considered and the calculated stresses are reported for the governing loads.

TABLE 3.9-2(f)
RECIRCULATION FLOW CONTROL VALVE
PRESSURE BOUNDARY MAXIMUM STRESSES

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS (psi)		ANALYTICALLY DETERMINED STRESS LEVEL (psi)*
1. Body (highest stress)	Per Section NB-3500 of the ASME Code, Section III	Primary:	17,380	8,860
		Primary* Secondary:	52,140	20,950
2. Top works housing (highest stress)	Per Section NB-3500 of the ASME Code, Section III	Primary:	17,380	5,730
		Primary* Secondary:	52,140	16,970
3. Pressure boundary studs	Per Section NB-3500 of the ASME Code, Section III	Peak:	81,090	30,270

*Based on seismic acceleration of 9g horizontal and 6g vertical.

TABLE 3.9-2(g)
SAFETY/RELIEF VALVES (MAIN STEAM)
SPRING-LOADED, DIRECT ACTING TYPE
ASME CODE, SECTION III, JULY 1974
(Including Addenda through Summer, 1976)

Topic	Method of Analysis	Analysis	Allowable Value	Calculated
1. Body inlet and outlet flange stresses Note, Topics 1 and 2 Design Pressures: P_d = 1375 psig (inlet) P_b = 625 psig (outlet) These are the max anticipated pressures under all operating conditions. Analyses include applied moments of M = 800,000 in.-lb (inlet) and M = 300,000 in.-lb (outlet) The analyses also include consideration of seismic, operational, and flow reaction forces. Since these safety/ relief valves are pipe-mounted nclude equipment, refer to the piping analysis for verification that the moments are not exceeded.	$S_H = \frac{fMo}{Lg_1^2B} + \frac{PB}{4g_o} < 1.5$ sm $S_R = \frac{4te/3+1}{Lt^2B} < 1.5$ Sm $S_T = \frac{YMo}{t^2B} - ZS_R < 1.5$ Sm where S_H = Longitudinal "Hub" wall stress, psi S_R = Radial "flange" (Body Base, Inlet) Stress, psi S_T = Tangential "flange" Stress, psi	(Uses same notation as codes) Body Material:ASME SA 352 LCB Inlet: Sm @ 585×F = 17,540 psi Outlet: Sm @ 500×F = 18,900 psi	1.5 Sm = 26,310 psi (inlet) = 28,350 psi (outlet)	Inlet: and S_H = 1.15 Sm = 0 (allowable) S_R = 0 (allowable) S_T = 0.98 Sm = 0.66 (allowable) Outlet: S_H = 1.21 Sm = 0.81 (allowable) S_R = 0.79 Sm = 0.53 (allowable) S_T = 0.49 Sm = 0.33 (allowable)

TABLE 3.9-2 (g) (Cont'd)

Topic	Method of Analysis	Analysis	Allowable Value	Calculated
2. Inlet and outlet stud area requirements	Total cross-sectional area shall exceed the greater of $Am_1 = \frac{Wm_1}{Sb}$, or $Am_2 = \frac{Wm_2}{Sb}$, or where Am_1 = total required bolt (stud) area for operating condition. Am_2 = total required bolt (stud) area for Gasket seating.	(Uses same notation as codes) Bolting Material: ASME: SA 193 Gr B7	Inlet: Am_1 ($>Am_2$) = 12.45 in.^{2} Outlet: Am_1 ($>Am_2$) = 4.65 in.^{2}	Inlet: Ab (actual area) = 1.52 A (required min) Outlet: Am (actual area) = 1.84 (required min)

TABLE 3.9-2 (g) (Cont'd)

Topic	Method of Analysis	Analysis	Allowable Value	Calculated
3. Nozzle Wall thickness	1 Minimum Wall Thickness Criterion: $t_{min} < t_A$ where t_{min} = minimum calculated thickness requirement, including corrosion allowance: t_A = Actual nozzle wall thickness. (NOTE: This t_{min} is t| per notation of the codes).	<u>Section near nozzle base</u>: $t\|_e < t\|_e$ (actual) <u>Nozzle mid-section</u>: $t\|_c < t\|_c$ (actual) <u>Thin section near valve seat</u>: $t\|_b < t\|_b$ (actual) <u>Thinnest section at nozzle tip - just below valve seat</u>: $t\|_a < t\|_a$ (actual) Nozzle Material: ASME SA 350 LF2	$t\|_e$ = 0.84 in. $t\|_c$ = 0.81 in. $t\|_b$ = 0.79 in. $t\|_a$ = 0.20 in. Actual thickness greater than t| at the section under consideration	$t\|_e$ (actual) $t\|_c$ (actual) = 1.54 $t\|_e$ $t\|_b$ (actual) = 1.012 $t\|$ $t\|_a$ (actual) = 1.68 $t\|_a$
(Refer to Section 3.9.1.1.9 for thermal transients information.)	2. Cyclic Rating: <u>Thermal</u>: $It = \sum \frac{Nri}{Ni}$ <u>Fatigue</u> Na ≥ 2,000 cycles, as based on Sa, where Sa is defined as the larger of $Sp_1 = (2/3)Q_p + P_{eb} + Q_{T_3} + 1.30_{T_1}$ or $Sp_2 = 0.4Q_p + \frac{K}{2}\left(P_{eb} + 2Q_{T_3}\right)$ where Sp_1 = Fatigue stress intensity at inside surface of crotch, psi. Sp_2 = Fatigue stress intensity at inside surface of crotch, psi.	$It = \sum \frac{Nri}{Ni}$ (i = 1, 2, 3, 4, and 5 Na ≥ 2,000 cycles, as based on Sa, where Sa = Sp_1 (> SP_2) (Uses same notation as codes)	It (max) ≤ 1.0 Na ≥ 2,000 cycles	It = 0.00138 (=0.00138 It (ma Na (based on Sa = Sp_2) = 400,000 cycles; ∴ criterion satisfied

TABLE 3.9-2 (g) (Cont'd)

	Topic	Method of Analysis	Analysis	Allowable Value	Calculated
4.	Bonnet Flange Strength	Flange treated as a loose type flange without hub: $S_R = \pm \frac{6\,M_p}{t^2(3.14\,C - nD)}$ $S_T = \pm \frac{5.46\,M_P}{B\,t^2}\ 0.318\ \frac{(C-B}{(C+B} + \frac{2h_c)}{C+A} + r_B - \frac{E\,A\,t}{B}$ S_R = Radial "Flange" Stress, psi S_T = Tangential "Flange" stress, psi.	(Uses same notation as code) Bonnet Material: ASME SA 352LCB Sm at 500×F = 18,900 psi	1.5 Sm (for max S_H, S_R, and S_T) = 28,350 psi	S_R = 1.35 Sm = 0.9 (allowable) S_T = 0.53 Sm = 0.35 (allowable) (max S_T @ back face of flange)
5.	Bonnet bolting area requirements	Total cross-sectional area shall exceed the greater of: $Am_1 = \frac{Wm_1}{Sb}$, or $Am_2 = \frac{Wm_2}{Sa}$ where Am_1 = Total required bolt (stud) area for operating condition. Am_2 = Total required bolt (stud) area for gasket seating.	$Am_1 = \frac{Wm_1}{Sb}$ $Am_2 = \frac{Wm_2}{Sa}$ where Am (required Minimum) is the greater of Am_1 and Am_2; and Ab (actual bolt area) must exceed Am Body to Bonnet Bolting Material: ASME SA 193 Gr B7	Am_1 (> Am_2) = 7.399 in.^{2}	A_b (actual area) = 1.34A_m (required minimum)

TABLE 3.9-2 (g) (Cont'd)

Topic	Method of Analysis	Analysis	Allowable Value	Calculated
6. Disc	The disc stress is calculated based on treating the disc as a flat circular plate, edges supported, uniform load over area with radius ro; reference Bach's Formulas, Machinery's Handbook, 15th Ed., page 414.			

TABLE 3.9-2 (g) (Cont'd)

Topic	Method of Analysis	Crosby 6-R-10 Analysis	Allowable Value	Calculated
Disc (Continuation)	From the reference, $$t = 1.2 \frac{W - \frac{2 r_o}{3 R}}{S}$$			
	W is based on p = 1375 psi under the disc.	W = 27,430 lbs. r_o = 0.785 inch R = 2.48 inches Disc Material: ASME SA 351 CF3A Temperature: 585°F Sm (585°F) = 18,235 psi Allowable stress is 1.5 Sm. This is the value "S" in the above formula. [1.5 Sm = 27,353 psi°	t (minimum allowable) = 1.067 inches	Actual t min =1.068 in = 1.0009 (require minimum
8. Seismic Capability	Stress analysis uses $F_{vertical}$ = (mass of valve) x (4.5g), and $F_{horizontal}$ = (mass of valve) x (6.5g), with 800,000 in-lb and 300,000 in-lb applied at the inlet and outlet, respectively. Actual capability verified by test (with the moments concurrently applied), and exceeds these values.			

TABLE 3.9-2(h)
MAIN STEAM ISOLATION VALVE
DESIGN OF PRESSURE RETAINING PARTS - ASME B&PV CODE SECT III 1974

ITEM NUMBER	COMPONENT/ LOAD TYPE/ STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE VALUE	DESIGN/ CALCULATED VALUE	RATIO CALCULATED ALLOWABLE
1.0	Body and bonnet				
1.1	Loads:				
	Design pressure	GE system specification	N/A	1375 psi	N/A
	Design temperature	GE system specification	N/A	586°F	N/A
	Pipe reaction loads				
1.2	Pressure rating	Table NB 3542.1-2	P_R = 575 lbf	P_R = 575 lbf	N/A
1.3	Minimum wall thickness	Paragraph NB 3542	t ≤ 1.643 in	t_m = 1.88 in	N/A
1.4	Primary membrane stress	Paragraph NB-3545.1 (500° F)	P_m = 19,400 psi	P_m = 11,317 psi	0.58
1.5	Secondary stress due to pipe reaction	Paragraph NB 3545.2 (b) (1)	P_{ed}, P_{eb}, and P_{et}	P_{ed} = 6,281 psi	0.23
			≤ 1.55m (500° F)	P_{ed} = 12,851 psi	0.48
			1.5 S_m = 26,700	P_{et} = 12,121 psi	0.45

TABLE 3.9-2 (h) (Cont'd)

ITEM NUMBER	COMPONENT/ LOAD TYPE/ STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE VALUE	DESIGN/ CALCULATED VALUE	RATIO CALCULATED ALLOWABLE
1.6	Primary plus secondary stress due to internal pressure	Paragraph NB-3545.2 (a) (1)		Q_b = 32,238 psi	
1.7	Thermal secondary	Paragraph NB-3545.2 (c)		Q_{T_2} = 347 psi	
1.8	Range of primary plus secondary stress at crotch region	Paragraph NB-3545.2	S_n = 58,200 psi	S_n = 39,213 psi	0.67
1.9	Body shape rule	Paragraph NB-3554	$r_2 \geq 0.5625$ in	r_2 = 0.94 in	
	- Radius at crotch	Paragraph NB-3554.1 (a)			
	- Corner radius	Paragraph NB-3554.1 (b)	$r_4 \leq 0.94$ in	r_4 = 0.88 in	
	- Longitudinal curvature	Paragraph NB-3544.6	>> 0.061 1/in	= 0.125 1/in	
	- No flat walls	Paragraph NB-3544.7			
	- Minimum wall at weld ends	Paragraph NB-3544.8	>> 1.444 in	= 1.587 ins	

TABLE 3.9-2 (h) (Cont'd)

ITEM NUMBER	COMPONENT/ LOAD TYPE/ STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE VALUE	DESIGN/ CALCULATED VALUE
1.10	Cyclic requirement for fatigue analysis	Paragraph NB-3545.3	$N_a \geq 2{,}000$ cycles	$N_a = 25{,}000$ cycles
1.11	Cumulative usafe factor requirements for fatigue analysis	Paragraph NB-3550	$I_t \leq 1.0$	$I_t = 0.017$
2.0	Body flange/bonnet			
2.1	Loads: (1) Design pressure (2) Design temperature (3) External moments due to dynamic loads that include SSE accelerations.	 1375 psi 586° F		

TABLE 3.9-2 (h) (Cont'd)

ITEM NUMBER	COMPONENT/ LOAD TYPE/ STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE VALUE	DESIGN/ CALCULATED VALUE	RATIO CALCULATED ALLOWABLE
2.2	Body flange stresses	Paragraph NB-3647.1		P_{FD} = 11,781 psig	
	Longitudinal hub stress		$S_H \leq 1.5\ S_m$ = 26,700 psi	S_H = 21,961 psi	0.82
	Radial flange stress		$S_R \leq 1.5\ S_m$ = 26,700 psi	S_R = 16,794 psi	0.62
	Tangential flange		$S_T \leq 1.5\ S_m$ = 26,700 psi	S_T = 8,900 psi	0.33
2.3	Bonnet thickness	Paragraph NB-3646 corrosion allowance = 0.12 IN	$t_m \geq 5.60$ in	t_m = 5.75 in	
2.4	Bonnet reinforcement	Paragraph NB-3646 (e)	Area $\geq 7.842\ in^2$	Area = $10.72\ in^2$	
3.0	Bonnet to body bolting Loads: (1) Design pressure (2) Design temperature (3) External moments due to dynamic loads which include SEE accelerations (4) Actuator operational loads.	Appendix XI	Ab $\geq 38.69\ in^2$	Ab = $50.69\ in^2$	1.31

TABLE 3.9-2 (h) (Cont'd)

ITEM NUMBER	COMPONENT/ LOAD TYPE/ STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE VALUE	DESIGN/ CALCULATED VALUE	RATIO CALCULATED ALLOWABLE
4.0	Valve poppet				
4.1	Loads: (1) In-line pressure loads				
4.2	Maximum stress	Roark's formulas for stress and strain 3rd edition, Cases 13, 14, 21, and 22.	$S_t \leq$ 17,800 psi $S_s \leq$ 10,680 psi	S_t = 14,007 psi S_s = 4,340 psi	0.79 0.40
5.0	Valve stem				
5.1	Loads: (1) Axial loads				
5.2	Under-cut thread stress	Industry standards	$S_t \leq$ 26,280 psi	S_t = 18,519 psi	0.70
5.3	Thread shear stress		$S_s \leq$ 26,280 psi	S_s = 7,177 psi	0.27
5.4	Buckling force		F > 46,963 lbs	F = 61,659 lbs	1.31

TABLE 3.9-2 (i)
RECIRCULATION PUMP SUMMARY LOAD CLASSIFICATION AND LIMIT CRITERIA, RECIRCULATION PUMP CASE

Loading Condition ASME Sec. III	LOAD	COMBINATION Mechanial Pressure	Criteria (ASME Sect. III Loads	Location NB-3220	Highest Calc. Stress/	Allowable Usage Fact.	Ratio Act./ A1
Design (NB-3112)	Design Pressure (1650 psig)	Pump Thrust Deadweight Nozzle Loads Gasket Seating Load	Fig. NB-3221-1 p| ≤ 1.0 S| $P_L+P_b \le 1.5 S_m$	Suction Transition	27250/ .0008	1.5 Sm = 28687	.95
Normal (NB-3113.1) and Upset (NB-3113.2)	Most Severe Normal/ Upset Pressure (1313 psig)	Deadweight Nozzle Loads Thermal-Transients 1/2 SSE Upset Only	Fig. NB-3222-1 $P_L+P_b+P_e+Q \le 3.0 S_m$ $P_e \le 3.0 S_m$	Casing Wall	57222/ .0038	3 Sm = 57375	.99
Emergency (NB-3113.3)	Most Severe Emergency Pressure (1796 psig)	Deadweight Nozzle Loads Pump Thrust Gasket Seating Load*	Fig. NB-3224-1 P| ≤ (1.2 S| or S_y) P_L ≤ (1.8 S| or 1.5 S_y) $P_L + P_b$ ≤ (1.8 S| or 1.5 S_y)	Suction Nozzle	30404/ .002	1.8 Sm = 34425	.88
Faulted (NB-3113.4)	Most Severe Faulted (1313 psig)	Deadweight Nozzle Loads SSE Pump Thrust Gasket Seating Load*	Table F-1322.2-1 P| ≤2.4 S| or 0.7 S_u + P_L ≤1.5 (2.4 S| or 0.7 S_u) P_L +Pb ≤ 1.5 (2.4 S| or 0.75 S_u)	Crotch	34162/ .0008	73500	.46

NOTE: The recirculation pumps are designed in accordance with Section III of the ASME code. The results of the complete analysis are shown in the Recirculation Pump Stress Report. These results are within the code allowable limits.

TABLE 3-9.2(j)
REACTOR RECIRCULATION SYSTEM
20" SUCTION GATE VALVE

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
1.0	Body and Bonnet				
1.1	Loads: Design Pressure Design Temperature	 System Requirement System Requirement	 1250 psi 575°F	 1250 psi 575°F	 N/A N/A
1.2	Pressure Rating psi	ASME Section III [1],Figure NB-3545.1-2	Pr=734.96 psi	Pr=734.96 psi	N/A
1.3	Minimum wall thickness, inches	ASME Section III [1],Para NB-3542	t| = 1.566 inches	t| = 1.566 min inches	N/A
1.4	Primary membrane Stress, psi	ASME Section III [1],Para NB-3545.1	P|≤S| (800°F) = 19,600 psi	P|=10,695 psi	0.54
1.5	Secondary stress due to pipe reaction	ASME Section III [1],Para NB-3545.2(b) (i)	Pe = Greatest value of Ped, Peb, and Pet ≤ 1.5 Sm (500°F) (1.5(19600) = 29,400 psi	Ped=8,298 psi Peb=22,059 psi Pet=21,211 psi Pe=Peb=22059 psi Pe=Peb=22059 psi	0.28 0.75 0.72 0.75

Note (1) : ASME Section III, 1971 Edition
(2) : Valve Differential Pressure 50 psiG

TABLE 3.9-2 (j) (Cont'd)

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
1.6	Primary plus secondary stress due to internal pressure	ASME Section III [1] ,Para NB-3545.2 (a) (1)	See paragraph 1.8	Qp=21,553 psi	
1.7	Thermal secondary stress	ASME Section III [1] , para NB-3545.2 (c)	See paragraph 1.8	Qt=3514 psi	
1.8	Range of primary plus secondary stress at crotch region.	ASME Section III [1] , para NB-3545.2	Sn≤3Sm (500°F) =3 (19600) = 58,800 psi	Sn=Qp+Pd+2Qt =32,059	0.54
1.9	Cycle Requirements for fatigue analysis	ASME Section III [1] , para NB-3545.3	Na≥2,000 cycles	Na=230,000 cycles	N/A
1.10	Usage factor requirements for fatigue analysis	ASME Section III [1] , para NB-3550	$I_t \leq 1.0$	$I_t = 0.0023$	N/A

TABLE 3.9-2 (j) (Cont'd)

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
2.0	Body to Bonnet Bolting				
2.1	Loads 1) Design pressure and 2) Temp., 3) Gasket loads 4) stem operational load, 5) Seismic load (safe shutdown earthquake)		-	-	-
2.2	Bolt area	ASME Section III [1], para NB-3647.1	$Ab \geq 23.73$ in^2 Sb=28,675 psi	Ab=28.1 in^2. Sb=25,398 psi	N/A 0.89
2.3	Body Flange Stresses	ASME Section III [1], para NB-3647.1	-	-	-
2.3.1	(1) Operating condition	ASME Section III [1], para NB-3647.1	$Sh \leq 1.5Sm(575°F)$ = 28,837 psi	SH=19,262 psi	0.67
			$SR \leq 1.5Sm(575°F)$ = 28,837 psi	SR=21,663 psi	0.75
			$ST \leq 1.5Sm(575°F)$ = 28,837 psi	ST=1,240 psi	0.04

TABLE 3.9-2 (j) (Cont'd)

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
2.3.2	Gasket seating condition	ASME Section III [1], para NB-3647.1	SH≤1.5Sm(100°F) = 30,000 psi	SH = 25,318 psi	0.84
			SR≤1.5Sm(100°F) = 30,000 psi	SR = 29,385 psi	0.98
			ST≤1.5Sm(100°F) = 30,000 psi	ST = 1,707 psi	0.05
2.4	Bonnet Flange stresses	ASME Section III [1], para NB-3647.1	-	-	-
2.4.1	Operating condition	ASME Section III [1], para NB-3647.1	SH≤1.5Sm(575°F) = 28,837 psi	SH = 18,963 psi	0.66
			SR≤1.5Sm(575°F) = 28,837 psi	SR = 17,032 psi	0.59
			ST≤1.5Sm(575°F) = 28,837 psi	ST = 2,241 psi	0.08

TABLE 3.9-2 (j) (Cont'd)

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
2.4.2	Gasket seating condition	ASME Section III [1] , para NB-3647.1	SH≤1.5Sm(100°F) =30,000 psi	SH=25,804 psi	0.86
			SR≤1.5Sm(100°F) =30,000 psi	SR=24,003 psi	0.80
			ST≤1.5Sm(100°F) =30,000 psi	ST=3293 psi	0.11
3.	Stresses in Stem				
3.1	Loads 1)operator thrust and 2)torque	-	-	-	-
3.2	Stem thrust stress	Calculate stress due to operator thrust in critical cross section	ST≤Sm =42,275 psi	Sr=4,289 psi	0.10
3.3	Stem torque stress	Calculate shear stress due to operator torque in critical cross section	Ss≤0.6Sm =42,275 psi	Ss=3,288 psi	0.07

TABLE 3.9-2 (j) (Cont'd)

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
3.4	Buckling on stem	Calculate slenderness ratio if greater than 30, calculate allowable load from Rankine's formula using safety factor of 4	Max. Allowable load=84,310 lbs Actual load on stem =15,577 lbs	Slenderness Ratio=40.96 0.18 Therefore, no buckling.	N/A
4.	Disc Analysis				
4.1	Loads: Maximum differential pressure	-	-	-	-
4.2	Maximum stress in the disc.	ASME Section III[1] ,para NB-3215 and ASME Section III[1] ,para NB-3221.3	Smax≤1.5Sm(575°F) =27,487 psi	Max stress =23,661 psi	0.86

TABLE 3.9-2 (j) (Cont'd)

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
5.	Yoke and Yoke Connections				
5.1	Loads & Stem Operational Loads	Calculate Stresses in the yoke and yoke connections to acceptable structural analysis methods.	-	-	-
5.2	Tensile stress in yoke leg bolts		Smax≤Sm(185°F) =32,960 psi	Smax=9,186 psi	0.28
5.3	Bending stress of yoke legs	-	Sb≤1.8Sm(185°F) =39,798 psi	Sb=17,133 psi	0.43

TABLE 3.9-2 (j) (Cont'd)

RECIRCULATION SUCTION GATE VALVE

Data deleted intentionally

TABLE 3.9-2 (j)
REACTOR RECIRCULATION SYSTEM
20" DISCHARGE GATE VALVE

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
1.	Body and Bonnet				
1.1	Loads:				
	Design pressure	System requirement	1650 psi	N/A	N/A
	Design temperature	System requirement	575°F	N/A	N/A
	Pipe reaction				
	Thermal effects	Not specified	N/A	N/A	N/A
1.2	Pressure rating, psi figure NB 3545.1-2	ASME Section III [1]	Pr=969.68 psi	Pr=969.68 psi	N/A
1.3	Minimum wall thickness, inches	ASME Section III [1] para NB-3542	t (nominal) =2.0937 inches	tm=2.077 min inches	N/A
1.4	Primary membrane stress, psi	ASME Section III [1] para NB-3545.1	Pm£Sm (500°F) =19600 psi	Pm=11,870 psi	0.60
1.5	Secondary stress due to pipe reaction	ASME Section III [1] ,para NB-3545.2 (b) (1) (S=30,000 psi)	Pe=greatest value of Ped, Peb and Pet ≤1.5Sm (500°F) 1.5 (19,600) =29,400 psi	Ped=6,861 psi Peb=15,342 psi Pet=15,350 psi Pe=Pet=15,350 psi	0.23 0.52 0.52 0.52

Note: (1) ASME Section III 1971 Edition
(2) Valve Differential Pressure 400 psig

TABLE 3.9-2 (j) (Cont'd)

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
1.6	Primary plus secondary stress due to internal pressure	ASME Section III [1] , para NB-3545.2 (a) (1)	Sn≤3Sm (500°F) =3 (19600) =58,800 psi	Qp=22,076 psi	0.37
1.7	Thermal secondary stress	ASME Section III [1] , para NB-3545.2 (c)	Sn≤3Sm (500°F) =3 (19600) =58,800 psi	Qt=2,895 psi	0.05
1.8	Sum of primary plus secondary stress	ASME Section III [1] , para NB-3545.2	Sn≤3Sm (500°F) =3 (19600) =58,800 psi	Sn=Qp+Pe+2QT =29,727 psi	0.50
1.9	Fatigue requirements	ASME Section III [1] , para NB-3545.3	Na≥2000 cycles	Na=10^6 cycles	N/A
1.10	Cycle rating	ASME Section III [1] , para NB-3550	It≤1.0	It=0.0012	N/A

TABLE 3.9-2 (j) (Cont'd)

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
2.0	Body to Bonnet Bolting				
2.1	Loads: Design pressure and temperature, gasket loads, stem operational load, seismic load (design basis earthquake)	ASME Section III [1], Para NB-3647.1	-	-	-
2.2	Bolt Area	ASME Section III [1], para NB-3647.1	$Ab \geq 34.87\ in^2$ $Sb \leq 28{,}675$ psi	$Ab = 40.32\ in^2$ Sb=25,478 psi	N/A 0.89
2.3	Body Flange stresses	ASME Section III [1], para NB-3647.1	-	-	-
2.3.1	(i) Operating conditions	ASME Section III [1], para NB-3647.1	SH≤1.5Sm (575°F) =28,837 psi	SH=16,098 psi	0.56
			SR≤1.5Sm (575°F) =28,837 psi	SR=12,443 psi	0.43
			ST≤1.5Sm (575°F) =28,837 psi	ST=1,975 psi	0.07

TABLE 3.9-2 (j) (Cont'd)

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
2.3.2	(ii) Gasket Seating condition	ASME Section III [1], Para NB-3647.1	SH≤1.5Sm(100°F) =30,000 psi	SH=21,963 psi	0.73
			SR≤1.5Sm(100°F) =30,000 psi	SR=17,849 psi	0.59
			ST≤1.5Sm(100°F) =30,000 psi	ST=2,809 psi	0.09
2.4	Bonnet Flange Stresses				
2.4.1	(i) Operating condition	ASME Section III [1], para NB-3647.1	SH≤1.5Sm(575°F) =28,837 psi	SH=17,875 psi	0.62
			SR≤1.5Sm(575°F) =28,837 psi	SR=11,850 psi	0.41
			ST≤1.5Sm(575°F) =28,837 psi	ST=3,160 psi	0.11

TABLE 3.9-2 (j) (Cont'd)

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
2.4.2	(ii) Gasket seating condition	ASME Section III [1] ,para NB-3647.1	SH≤1.5Sm (100°F) =30,000 psi	SH=20,731 psi	0.69
			SR≤1.5Sm (100°F) =30,000 psi	SR=13,632 psi	0.45
			ST≤1.5Sm (100°F) =30,000 psi	ST=3,957 psi	0.13
3.	Stress in stem				
3.1	Loads: Operator thrust and torque	-	-	-	-
3.2	Stem thrust stress	Calculate stress due to operator thrust in critical cross section	ST<Sm	ST=8,443 psi =42,275 psi	0.20
3.3	Stem Torque stress	Calculate shear stress due to operator torque in critical cross section	Ss≤0.6Sm =25,365	Ss=6,471 psi	0.25

TABLE 3.9-2 (j) (Cont'd)

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
3.4	Buckling on Stem	Calculate slenderness ratio. If greater than 30, calculate allowable load from Rankine's formula using safety factor of 4.	Max allowable load=53,803 lbs	Slenderness ratio=68.9 Actual load =30,623 lbs Therefore, no buckling.	N/A 0.57
4.	Disc Analysis				
4.1	Loads: Maximum differential pressure	-	-	-	-
4.2	Maximum stress in the disc.	ASME Section III [1] ,para NB-3215 and ASME Section III [1] ,para NB-3221.3	Smax≤1.5Sm(575°F) =27,487 psi	Max Stress =24850 psi	0.90

TABLE 3.9-2 (j) (Cont'd)

Paragraph Number	Component/Load/ Stress Type	Design Procedure	Allowable Limit	Design/Calculated Value	Ratio (Calculated Allowed)
5.	Yoke and Yoke Connections				
5.1	Loads & Stem operational Loads	Calculate stresses in the yoke and yoke connections to acceptable structural analysis methods.			
5.2	Tensile stress in yoke leg bolts	-	Smax≤Sm(185°F) =32,960 psi	Smax=18,616 psi	0.56
5.3	Bending stress of yoke legs	-	Sb≤1.8Sm(185°F) =39,798 psi	Smax=34,860 psi	0.88

Note: (1) Appropriate loading combinations of Table 3.9-2 were considered and the calculated stresses are reported for the governing loads.

TABLE 3.9-2 (j) (Cont'd)

RECIRCULATION DISCHARGE GATE VALVE

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TABLE 3.9-2 (k)
ASME CODE CLASS III SAFETY/RELIEF VALVE
DISCHARGE PIPING SYSTEM - HIGHEST STRESS SUMMARY

ACCEPTANCE CRITERIA	LIMITING STRESS TYPE	CALCULATED (1) STRESS	ALLOWABLE LIMITS	RATIO CALCULATED/ ALLOWABLE	LOADING	IDENTIFICATION OF LOCATION OF HIGHEST STRESS POINTS
ASME B&PV Code Section III, ND-3600						
Design condition: EQ 8 $\leq$ 1.0 S_h	Sustained loads	4518 psi	22500 psi	0.201	1. Pressure 2. Weight	MS Line C First node after SRV valve
Service Level A&B (Normal & Upset Condition) EQ 9 $\leq$ 1.2 S_h	Occasional loads	9730 psi	27000 psi	0.360	1. Pressure 2. Weight 3. OBE 4. SRV	MS Line A, Elbow
EQ 10 $\leq$ S_a	Thermal Expansion	13884 psi	22500 psi	0.617	1. OBE 2. SRV 3. Thermal	MS Line C, Elbow
EQ 11 $\leq$ $S_a + S_h$	Sustained and Thermal Expansion	18477 psi	37500 psi	0.493	1. Pressure 2. Weight 3. OBE 4. SRV 5. Thermal	MS Line C, Elbow

Note: Piping within the scope of this table has been evaluated for the effects of extended power uprate (EPU) and satisfy the applicable Code requirements. A summary of the Reactor Coolant Piping EPU evaluation is summarized in PUSAR NEDC-32989P.

TABLE 3.9-2 (k)
ASME CODE CLASS III SAFETY/RELIEF VALVE
DISCHARGE PIPING SYSTEM - HIGHEST STRESS SUMMARY

ACCEPTANCE CRITERIA	LIMITING STRESS TYPE	CALCULATED (1) STRESS	ALLOWABLE LIMITS	RATIO CALCULATED/ ALLOWABLE	LOADING	IDENTIFICATION OF LOCATION OF HIGHEST STRESS POINTS
Service Level C (Emergency Condition) EQ 9 $\leq$ 1.8 S_h	Primary loads	9726 psi	33750 psi	0.288	1. Pressure 2. Weight 3. SRV 4. LOCA	MS Line A Elbow Joint
Service Level D (Faulted Condition) ASME Code Case EQ 9 $\leq$ 2.4 S_h	Primary loads	13594 psi	45000 psi	0.302	1. Pressure 2. Weight 3. SSE 4. Annulus Pressurization	MS Line A Elbow Joint

Note: (1) Appropriate loading combinations of Table 3.9-2 were considered and the calculated stresses are reported for the governing loads.

TABLE 3.9-2 (I)
STANDBY LIQUID CONTROL PUMP

CRITICAL/LOADING		COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
Based on ASME B&PV Code Section III.					
Pressure boundary parts:					
1. Fluid cylinder - SA182-F304,	S_y = 30,000 psi				
2. Discharge valve stop and cylinder head extension SA 479-304,	S_y = 30,000 psi				
3. Discharge valve cover, cylinder head & stuffing box flange plate, SA 240-304,	S_y = 30,000 psi				
4. Stuffing box gland, SA 564-630	S_y = 115,000 psi				
5. Studs, SA 192B7,	S_y = 105,000 psi				
6. Dowel pins (2) alignment, SAE 4140,	S_A = 23,400 psi				
7. Studs, cylinder tie, SA 193-B7,	S_A = 25,000 psi				
8. Pump holddown bolts, SAE GR.8,	T_A = 30,000 psi Q_A = 37,500 psi				
9. Power frame, foot area, cast iron,	S_A = 15,000 psi				
10. Motor holddown bolts, SAE GR.8	T_A = 30,000 psi Q_A= 37,500 psi				
11. Motor frame foot area, cast iron,	S_A = 7,500 psi				
Normal & Upset Condition Loads:					
1. Design pressure		1. Fluid cylinder	General membrane	17,800	
2. Design temperature		2. Discharge valve stop	General membrane	17,800	See note (3)
3. Operating basis earthquake		3. Cylinder head extension	General membrane	17,800	
4. Nozzle loads (1)		4. Discharge valve cover	General membrane	17,800	
5. SRV discharge		5. Cylinder head	General membrane	17,800	
		6. Stuffing box flange plate	General membrane General membrane	17,800 17,800	
		7. Stuffing box gland	General membrane	35,000	
		8. Cylinder head studs	Tensile	25,000	
		9. Stuffing box studs	Tensile	25,000	

TABLE 3.9-2 (I) (Cont'd)

CRITICAL/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
Emergency Conditions:				
1. Design pressure	1. Fluid cylinder	General membrane	21,360	4,450
2. Design temperature	2. Discharge valve stop	General membrane	21,360	13,600
3. Dead weight	3. Cylinder head extension	General membrane	21,360	13,600
4. Thermal expansion	4. Discharge valve cover	General membrane	21,360	8,150
5. Nozzle loads	5. Cylinder head	General membrane	21,360	8,150
6. Safety relief valve discharge	6. Stuffing box flange plate	General membrane	21,360	10,390
7. LOCA	7. Stuffing box gland	General membrane	42,000	11,420
Faulted Conditions:				
1. Design pressure	1. Cylinder head studs	Tensile	25,000	18,820
2. Design temperature	2. Stuffing box studs	Tensile	25,000	24,750
3. Nozzle loads	3. Dowel pins (2)	Shear only (2)	23,400	19,430
4. Safety relief valve discharge	4. Studs, cylinder tie	Tensile (2)	25,000	8,685
5. LOCA	5. Pump holddown bolts	Shear	30,000	11,350
6. Safe shutdown earthquake	6. Pump holddown bolts	Tensile	37,500	17,680
7. Dead weight	7. Power frame-foot area	Shear	15,000	1,850
8. Thermal expansion	8. Power frame-foot area	Tensile	15,000	11,390
	9. Motor holddown bolts	Shear	30,000	3,020
	10. Motor holddown bolts	Tensile	37,500	5,290
	11. Motor holddown bolts	Shear	7,500	2,550
	12. Motor holddown bolts	Tensile	7,500	5,100
Faulted Condition Dynamic Loads:				
1. SRV	SLC Pump Assembly	Acceleration	1.75g horizontal 0.83g	
2. SSE			1.75g vertical 0.41g	
3. LOCA				

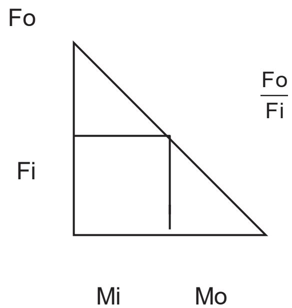
TABLE 3.9-2 (I) (Cont'd)

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE LOADS (lbf, ft-lbf)	CALCULATED (lbf, ft-lbf)

<u>Nozzle Load Definition</u>:

Units Forces - lbs
Moments - ft - lbs

Allowable combination of forces and moments are as follows:



$$\frac{Fo}{Fi} + \frac{Mi}{Mo} \leq 1$$

Where:

F_1 = The largest absolute value of the three actual external orthogonal forces (F_x, F_y, F_z) be imposed by the interface pipe, and,

M_i = The largest absolute value of the three actual internal orthogonal moments (M_x, M_y, M_z) permitted from the pipe when they are combined simultaneously from the pipe when they are combined simultaneously for a specific condition.

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE LOADS (lbf, ft-lbf)	CALCULATED (lbf, ft-lbf)
			SUCTION:	
<u>Normal and Upset Condition Loads</u>: 1. Design pressure		F_o = Allowable value of f_i when all moments are zero.	F_o =770	F = 235
2. Design temperature				
3. Operating basis earthquake		M_o = Allowable value of M_1 when all moments are zero.	M_o =490	M = 239
			DISCHARGE:	
4. Nozzle loads				
5. SRV discharge			F_o =370	F = 124
6. Dead weight			M_o = 110	M = 67
7. Thermal expansion				

TABLE 3.9-2 (I) (Cont'd)

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE LOADS (lbf, ft-lbf)	CALCULATED (lbf, ft-lbf)
Emergency or Faulted Condition Loads:			SUCTION:	
			F_o = 920	F = 274
1. Design pressure			M_o= 590	M = 291
2. Design temperature				
3. Nozzle loads			DISCHARGE:	
4. SRV discharge				
5. LOCA			F_o = 440	F = 148
6. Safe shutdown earthquake			M_o= 130	M = 79

NOTES:

(1) Nozzle loads produce shear loads only.

(2) Dowel pins take all shear.

(3) Calculated stresses for emergency or faulted condition are less than the allowable stresses for the normal and upset condition loads, therefore the normal and upset condition is not evaluated.

(4) Operability: The sum of the plunges and rod assembly, pounds mass times 1.75g acceleration is much less than the thrust loads encountered during normal operating conditions. Therefore, the loads during the faulted condition have no significant effect on pump operability.

TABLE 3.9-2 (m)
STANDBY LIQUID CONTROL TANK

CRITERIA	METHOD OF ANALYSIS	ALLOWABLE STRESS OR MIN. THICKNESS REQD OR LOAD		ACTUAL STRESS OR MIN. THICKNESS REQD OR LOAD
1. Shell Thickness				
Loads: Normal & Upset Design Pressure and Temperature	Brownell & Young "Process Equipment Design" $t = \frac{PR}{SE - 0.6\,P}$	0.016 In		0.25 In
Stress Limit	ASME Section III	18,300 psi		1203 psi
2. Nozzle Loads				
Loads: Normal & Upset Design Pressure and Temperature	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits.	Fo (lb)	Mo (ft.lb)	F = 166
Overflow Nozzle		770	490	M = 129
Discharge Nozzle		770	490	
Loads: Faulted Dead Weight, Thermal Expansion and SSE Earthquake	The maximum moments due to pipe reaction and maximum forces shall not exceed the allowable limits.	Fo (lb)	Mo (ft.lb)	F = 179
Overflow Nozzle		925	590	M = 144
Discharge Nozzle		925	590	
3. Anchor Bolts	ASME Section III	10,000 psi		4221 psi
4. Dynamic Loads	Equivalent Static	1.75g horizontal		0.645g horizontal
SRV SSE LOCA		1.75g vertical		0.378g vertical

TABLE 3.9-2 (n)
ECCS PUMPS

RESIDUAL HEAT REMOVAL PUMP

LOCATION	LOADING CONDITION	CRITERIA	CALCULATED STRESS (PSI) OR ACTUAL THICKNESS (In.)	ALLOWABLE STRESS (PSI) OR MIN. THICKNESS
Discharge head shell	Faulted Condition Design pressure Nozzle loads Seismic load SRV, LOCA loads	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-27	30,660	31,500
Discharge head cover	Design pressure	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-34 UG-39 & UG-40	3"	2.75" dwg. min. 1.84" code min. required
Nozzle shell	Faulted Condition Design pressure Nozzle loads Seismic load SRV, LOCA loads	ASME Boiler & Pressure Vessel Code, Section VIII, Division 1, Para. UG-37	30,660 (Suction) 24,441 (Discharge)	31,500
Discharge pipe or suction pipe (maximum)	Faulted Condition Design pressure Nozzle loads	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-27	17,894 (Suction) (Discharge)	18,000
Discharge head bolting	Faulted Condition Design pressure Nozzle loads Seismic load SRV, LOCA loads	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	34,446	45,000
Motor bolting	Faulted Condition Seismic load SRV, LOCA loads	Bolting, Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	3,842	25,000

TABLE 3.9-2 (n) (Cont'd)

LOW PRESSURE CORE SPRAY PUMP

LOCATION	LOADING CONDITION	CRITERIA	CALCULATED STRESS (PSI) OR ACTUAL THICKNESS (In.)	ALLOWABLE STRESS (PSI) OR MIN. THICKNESS
	Faulted Condition			
Discharge head shell	Design pressure Nozzle loads Seismic load	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-27	18,253	21,000
Discharge head cover	Design pressure	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-34, UG-39 & UG-40	3"	2.25" dwg. min. 1.43" code min. required
	Faulted Condition			
Nozzle shell intersection	Design pressure Nozzle loads Seismic load	ASME Boiler & Pressure Vessel Code, Section VIII, Division 1, Para. UG-37	18,253 (Suction) 13,986 (Discharge)	31,500
	Faulted Condition			
Discharge pipe or suction pipe (maximum)	Design pressure Nozzle loads	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-27	7,883 (Suction) 16,633 (Discharge)	18,000
	Faulted Condition			
Discharge head bolting	Design pressure Nozzle loads Seismic load	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	31,567	37,500
	Faulted Condition			
Motor bolting	Seismic load	Bolting, Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	4,858	25,000

TABLE 3.9-2 (n) (Cont'd)

HIGH PRESSURE CORE SPRAY PUMP

LOCATION	LOADING CONDITION	CRITERIA	CALCULATED STRESS (PSI) OR ACTUAL THICKNESS (In.)	ALLOWABLE STRESS (PSI) OR MIN. THICKNESS
	Faulted Condition			
Discharge head shell	Design pressure Nozzle loads Seismic load	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-27	13,480 psi	31,500
Discharge head cover	Design pressure	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-34, UG-39 & UG-40	3"	2.75" code min. required
	Faulted Condition			
Nozzle shell intersection	Design pressure Nozzle loads Seismic load	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-37	9,844 (Suction) 13,480 (Discharge)	31,500 psi
	Faulted Condition			
Discharge pipe or suction pipe (maximum)	Design pressure Nozzle loads	ASME Boiler & Pressure Vessel Code, Section VIII Division 1, Para. UG-27	8,890 (Discharge)	18,000
	Faulted Condition			
Discharge head bolting	Design pressure Nozzle loads Seismic load	Bolting Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	21,331 psi	45,000 psi
	Faulted Condition			
Motor bolting	Seismic load	Bolting, Loads & Stresses per "Rules for Bolted Flange Connections" ASME Section VIII, App. II	5,991 psi	25,000 psi

NOTE: Operability demonstrated by analysis, vendor operator experience, and testing.

TABLE 3.9-2 (o)
CLINTON RHR HEAT EXCHANGER

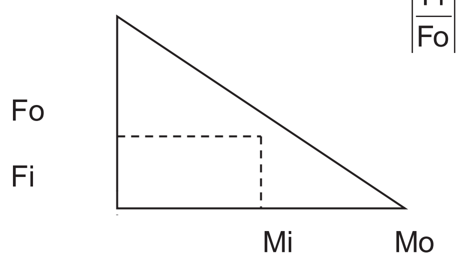
Loading	Criteria	Allowable Stress or Min. Thickness Reqd.	Calculated Stress or Thickness
1. Closure Bolting Loads: Normal and Upset Design pressure and temperature Design gasket load	Bolting loads and stresses calculated per "Rules for Bolted Flange Connections" ASME Section III, App XI		
Bolting Stress Limit	a. Shell to tube sheet bolts	25,000 psi	23,527 psi
Allowable working stress per ASME Section III	b. Channel cover bolts	25,000 psi	24,521 psi
2. Wall Thickness	Shell side ASME Section III, Class 2 and TEMA, Class C		
Design pressure and temperature	Tube Side ASME Section III, Class 3 and TEMA, Class C		
Stress Limit ASME Section III	a. Shell b. Shell cover c. Channel ring d. Tubes e. Channel cover f. Tube sheet	0.838 in. 0.827 in. 0.86 in. 7.42 in. 6.65 in.	0.88 in. 0.827 in. min. 0.88 in. 16 BWG 7.44 in. 6.75 in.

TABLE 3.9-2 (o) (Cont'd)

Loading	Criteria	Allowable Nozzle Forces and Moments Force in lb., Moment in ft-lb	Actual Nozzle Load
3. <u>Nozzle Loads</u>	The maximum moments due to pipe reaction and the maximum forces shall not exceed the allowable limits.	See below. * (a) * (b)	N_1: F = 3304 M = 8624 N_2: F = 3750 M = 10642
Design pressure and Temperature			
Dead Weight, thermal expansion safe shutdown earthquake	Primary stress small of 0.7 Su or 2.4 ASME Section III allowable		N_3: F =4942 M = 16836 N_4: F =3712 M = 17543

*(a) The following expression relates the allowable combination of forces and moments:

$$\left|\frac{Fi}{Fo}\right| + \left|\frac{Mi}{Mo}\right| \leq 1$$



Where:

Fi = The largest of the three actual external orthogonal forces (Fx, Fy, and Fz)
Mi = The largest of the three actual external orthogonal moments (Mx, My, and Mz)
Fo = The allowable value of Fi when all moments are zero
Mo = The allowable value of Mi when all forces are zero

TABLE 3.9-2 (o) (Cont'd)

*(b) Allowable limits (Design basis)

	N1	N2	N3	N4
Fx =	10,500 lb	10,500 lb	13,000 lb	13,000 lb
Fy =	10,500 lb	10,500 lb	13,000 lb	13,000 lb
Fz =	10,500 lb	10,500 lb	13,000 lb	13,000 lb
Mx =	32,000 ft-lb	32,000 ft-lb	46,000 ft-lb	46,000 ft-lb
My =	32,000 ft-lb	32,000 ft-lb	46,000 ft-lb	46,000 ft-lb
Mz =	32,000 ft-lb	32,000 ft-lb	46,000 ft-lb	46,000 ft-lb

TABLE 3.9-2 (o) (Cont'd)

	COMPONENT/LOADING	CRITERIA/LOCATION	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED (psi)	ACTUAL (psi)
4.	Support Brackets & Attachment Welds	Stress allowables as per ASME Section III Subsection NT (Upset Condition)		
	Loads: Faulted			
	Design pressure and temperature, dead weight, nozzle loads, safe shutdown earthquake, safety relief valve LOCA	a. Lower bracket welds		
		- Bending stress	14,438	10,344
		- Shear stress	21,000	6,674
		b. Upper bracket welds		
		- Bending stress	14,438	4,437
		- Shear stress	21,000	2,599
5.	Anchor Bolts	Stress allowable as per ASME III, Appendix XVII		
	Loads: Faulted	Lower support bolting		
	Design Pressure and temperature, dead weight, nozzle loads, SSE, SRV, LOCA	- Tension	57,500	11,483
		- Shear	23,700	3,749
			(upset allowable)	(faulted loads)

TABLE 3.9-2 (o) (Cont'd)

	COMPONENT/LOADING	CRITERIA/LOCATION	ALLOWABLE STRESS OR MINIMUM THICKNESS REQUIRED (psi)	ACTUAL (psi)
6.	Shell Adjacent to Support Brackets	Shell stress allowable as per ASME Section III Subsection NC		
	Loads: Faulted			
	Design pressure and temperature, dead weight, nozzle loads, safe shutdown earthquake, safety relief valve LOCA	a. Maximum principal stress adjacent to upper support	42,000	30,887
		b. Maximum principal stress adjacent to lower support	26,250	21,246
7.	Shell Away from Discontinuities	Stress allowable as per ASME Section III Subsection NC		
	Loads: Faulted			
	Design pressure and temperature, dead weight, nozzle, loads, SSE, SRV	Principal stress	35,000	17,849

TABLE 3.9-2 (p)
REACTOR WATER CLEANUP PUMP

Component	Loading Condition	Stress Criteria	Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
Suction Nozzle	Design pressure and design temperature	$S \leq Sa$	General membrane	14,000	5,100
Discharge Nozzle	Design pressure and design temperature	$S \leq Sa$	General membrane	14,000	4,040
Cover Bolting	Design pressure and design temperature	$S \leq Sa$	General membrane	29,300	26,300
Seal Gland Bolting	Design pressure and design temperature	$S \leq Sa$	General membrane	29,300	22,100
Seal Gland	Design pressure and design temperature	$S \leq Sa$	General membrane	13,800	4,285
Pump Cover	Design pressure and design temperature	$S \leq Sa$	General membrane	32,400	9,060
Pedestal Bolts (Tensile)	Pressure loads Thermal loads Nozzle load Seismic load Dead weight Torsional load	$S \leq Sa$	General membrane	51,537	11,734
Pedestal Bolts (Shear)	Pressure loads Thermal loads Nozzle loads Seismic loads Torsional load	$S \leq Sa$	Shear	18,855	6,810
Pedestal Bolts	Preload	$S \leq 0.9\ Sy$	Shear	135,000	68,000

TABLE 3.9-2 (p) (Cont'd)

Component	Loading Condition	Stress Criteria	Stress Type	Allowable Stress (PSI)	Calculated Stress (PSI)
Motor hold-down bolts (Shear)	Pressure loads Thermal loads Seismic loads Dead weight Torsional load	S ≤ Sa	General membrane	30,000	2,955
Motor hold-down Bolts (Tensile)	Pressure loads Thermal loads Seismic loads Dead weight Torsional load	S ≤ Sa	General membrane	37,500	5,036
Foundation Bolts (Tensile)	Pressure loads Thermal loads Seismic loads Dead weight Torsional load	General Electric Design Specifi-cation	General membrane	10,000	6,315
Foundation Bolts Shear	Pressure loads Thermal loads Seismic loads Dead weight Torsional load	General Electric Design Specifi-cation	(Shear)	10,000	8,801
Shaft Deflection at pump wear ring	Pressure loads Thermal loads Seismic loads Dead weight Torsional load	General Electric Design Specifi-cation	N/A	10.0 (mils)	3.93 (mils)
Shaft deflection at pump coupling	Pressure loads Thermal loads Seismic loads Dead weight Torsional load	General Electric Design Specifi-cation	N/A	5.0 (mils)	3.345 (mils)

Nomenclature: Sa = Allowable general membrane stress per ASME Section III, 1974 Edition
S = Calculated stress

TABLE 3.9-2 (q)
RCIC TURBINE

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
The highest stressed sections of the various components of the RCIC turbine assembly are identified. Allowable stresses are based on ASME B&PV Code, Section III, for:				
Pressure Boundary Castings SA216-NCB: S = 14,000 psi				
Pressure Boundary Boltings SA193-B7 S = 25,000 psi				
Alignment Dowel Pins: AIS14037, Rc28-35 r_a = 61,000 psi S_y = 106,000 psi				
Normal and upset condition loads:	Castings:			
	1) Stop valve	General membrane	14,000	See Note 1
1. Design pressure	2) Governor valve	General membrane	14,000	
2. Design temperature	3) Turbine inlet	Local bending	21,000	
3. Operating basis earthquake	4) Turbine cases	Local bending	21,000	
	Pressure containing bolts:	Tensile	25,000	
	Structure alignment pins:	Shear	61,000	
4. Insert nozzle loads				
5. Exhaust nozzle loads				

TABLE 3.9-2 (q) (Cont'd)

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
Emergency or Faulted Conditions:	Castings:			
1. Design pressure	1. Stop valve	General membrane	16,800	9,800
2. Design temperature	2. Governor valve	General membrane	16,800	13,300
3. Safe shutdown earthquake	3. Turbine inlet	local bending	25,200	25,300
	4. Turbine case	Local bending	25,200	18,000
	Pressure containing bolts	Tensile	25,000	20,100
	Structure alignment pins	Shear	61,000	46,800
4. Inlet nozzle loads				
5. Exhaust nozzle loads.				
Nozzle Load definition:				
allowable nozzle loads for the turbine assembly. The above calculated stresses assume these allowable nozzle loads have been satisfied.				
Normal & Upset Condition Loads:			Allow load criteria	
1. Design pressure				
2. Design temperature			Inlet:	Inlet:
3. Weight of structure				
4. Thermal expansion			$F = \frac{(3500 - M)}{3}$	No analysis (small piping)
5. Operating basis earthquake				
			Exhaust: $F = \frac{(7000 - M)}{3}$	Exhaust: F = 1,285 M = 1,759
			F = Resultant force (lbs) M = Resultant moment (Ft-lbs)	

TABLE 3.9-2 (q) (Cont'd)

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
<u>Emergency & Faulted Condition Loads</u>:				
1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion 5. Safe shutdown earthquake			Inlet: $F = \frac{(4200 - M)}{3}$ Exhaust: $F = \frac{(8400 - M)}{3}$ F = resultant force (lbs) M = resultant moment (ft. lbs)	Inlet: No analysis (small piping) Exhaust: F = 1319 M = 1804

NOTES:

(1) Calculated stresses for the emergency or faulted condition are lower than the allowable stresses for the normal plus upset condition, therefore the normal and upset condition is not evaluated.

(2) Operability: Analysis indicates that shaft deflection with faulted loads is 0.006 inch; which is fully acceptable; and maximum bearing load with faulted condition is 80% of allowable. Furthermore, as indicated in Paragraph 3.9.2.2.1.6.9, the turbine assembly has been seismically qualified via dynamic testing, enveloping the response spectra. This justification included demonstration of start-up and shutdown capabilities, as well as no load operability during seismic loading conditions.

TABLE 3.9-2 (r)
RCIC PUMP

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
Pressure boundary stress limits of the various components for the RCIC pump assembly are based on the ASME B&PV Code Section III, for pressure boundary parts at 140° F.				
1. Forged barrel, SA105 GR. II S_y = 36,000 psi				
2. End cover plates, SA105 GR. II S_y = 36,000 psi				
3. Nozzle connections, SA105 GR. II S_y = 36,000 psi				
4. Aligning pin, SA105 GR. II S_y = 36,000 psi				
5. Closure bolting, SA193-87 S_y = 105,000 psi				
6. Pump holddown bolting, SA325 S_y = 77,000 psi				
7. Taper pins, SA108 GR B1112, S_y = 75,000 psi				
Normal & Upset Condition Loads:				
1. Design pressure	1. Forged barrel	General membrane		
2. Design temperature	2. Nozzle reinforcement	General membrane		
3. Operating basis earthquake				
4. Suction nozzle loads	3. Alignment pin	Shear	See Note 1	
5. Discharge nozzle loads	4. Taper pins	Shear		
	5. Pump holddown bolts	Tensile		

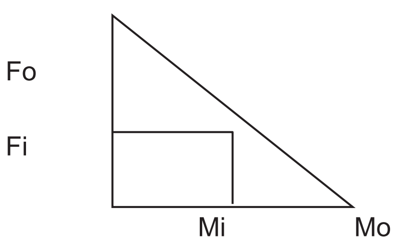
TABLE 3.9-2 (r) (Cont'd)

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
Emergency or Faulted Condition Loads:				
1. Design pressure	1. Forged barrel	General membrane	17,500	7,052
2. Design temperature 3. Safe shutdown earthquake 4. Suction nozzle loads	2. Nozzle reinforcement at barrel	General membrane	26,250	7,855
5. Discharge nozzle loads	3. Alignment pin	Shear	18,000	2,230
	4. Taper pins (baring housing)	Shear	15,000	2,280
	5. Pump holddown bolts	Tension	48,000	33,662

Nozzle Load Definition:

Units: Forces - lbs
Moments - ft-lbs

The allowable combinations of forces and moments are as follows:



$$\frac{Fi}{Fo} + \frac{Mi}{Mo} \leq 1$$

Where:

Fi = Largest absolute value of the three actual external orthogonal forces (Fx, Fy, Fz) that may be imposed by the interface pipe and,

Mi = Largest absolute value of the three actual external orthogonal moments (Mx, My, Mz) permitted from the interface pipe when they are combined simultaneously for a specific condition.

TABLE 3.9-2 (r) (Cont'd)

CRITERIA/LOADING	COMPONENT	LIMITING STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
Normal & Upset Condition Loads:				
1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion 5. Operating basis earthquake	Fo =	Allowable value of Fi when all moments are zero	Suction: Fo = 1,940 Mo = 2,460	 F = 529 M = 595
	Mo =	Allowable value of Mi when all forces are zero	Discharge: Fo = 3,715 Mo = 4,330	 F = 953 M = 2,157
Emergency or Faulted Condition Loads:				
1. Design pressure 2. Design temperature 3. Weight of structure 4. Thermal expansion 5. Safe shutdown earthquake			Suction: Fo = 2,325 Mo = 2,950	 F = 483 M = 545
			Discharge: Fo = 4,450 Mo = 5,200	 F = 1,107 M = 2,456

NOTES:

(1) Calculated stresses for emergency or faulted condition are less than the allowable for normal plus upset condition, therefore the normal and upset condition is not evaluated.

(2) Operability: static analysis for emergency or faulted condition show that the maximum shaft deflection is 0.004 in. with 0.005 in. allowable, shaft stresses are 597 psi with 32,000 psi allowable, and bearing loads, of drive and 376 lbs, with 7,670 lbs allowable and thrust end 1,323 lbs with 17,200 lbs allowable.

TABLE 3.9-2(s)
REACTOR REFUELING AND SERVICING EQUIPMENT

FUEL STORAGE RACKS

ACCEPTANCE CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
The allowable primary bending stress is based on ASME Section III for ASTM B221 6061-T6 aluminum alloy				
S_o = 38,000 psi				
S_y = 35,000 psi				
For normal condition:	For normal condition:			
S_{limit} = .49 S_y	1. Normal operating loads	Axial + bending	17,300	12,030
For emergency condition:	For emergency condition:			
S_{limit} = .66 S_y	1. Normal operating loads 2. Operating basis earthquake	Axial + bending	23,000	20,133
For faulted condition:	For faulted condition:			
S_{limit} = 1.0 S_y	1. Normal operating loads 2. Safe shutdown earthquake	Axial + bending	35,000	34,930

TABLE 3.9-2 (s) (Cont'd)

REFUELING PLATFORM

ACCEPTANCE CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
The allowable axial load stress is based on AISC Part 5 Section 1.5 for ASTM A36 structural steel for type:				
F_u = 58,000 psi				
F_y = 36,000 psi				
For normal condition:	For normal condition:			
$S_{limit} = 0.66 F_y$	1. Static loads	Axial + bending	23,760	1,886
For upset condition:	For upset condition:			
$S_{limit} = 0.88 F_y$	1. Static loads 2 SRV 3. OBE	Axial + bending	31,680	23,618
For faulted condition:	For faulted condition:			
$S_{limit} = 0.7 F_u$	1. Static loads 2. SSE 3. SRV 4. LOCA	Axial + bending	40,600	28,395

TABLE 3.9-2 (s) (Cont'd)

FUEL PREPARATION MACHINE

ACCEPTANCE CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
The allowable for axial plus bending stresses is based on ASME Code Section III ASTM A-157 for Type 3 annealed stainless Steel				
F_y = 30,000 psi				
F_u = 75,000 psi				
S_m = 17,800 psi				
For normal condition:	For normal condition:			
$S_{limit} = 1.0\ S_m$	1. Static loads	Axial + bending	17,800	16,030
For emergency condition:	For emergency condition:			
$S_{limit} = 0.8\ F_y$	1. Static loads 2. OBE 3. SRV	Axial + bending	24,000	16,148
For faulted condition:	For faulted condition:			
$S_{limit} = 1.2\ F_y$	1. Static loads 2. SSE 3. RV 4. LOCA	Axial + bending	36,000	16,132

TABLE 3.9-2 (s) (Cont'd)

INCLINED FUEL TRANSFER TUBE

ACCEPTANCE CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
The allowable for axial plus bending loads are based on for type				
F_y = 27,450 psi				
For normal condition:	For normal condition:			
$S_{limit} = 1.0\ F_y$	1. Normal loads	Axial + bending	27,450	7,750
For emergency condition:	For emergency condition:			
$S_{limit} = 1.5\ F_y$	1. Normal loads 2. OBE 3. SRV_{ALL}	Axial + bending	41,175	20,413
For faulted condition:	For faulted condition:			
$S_{limit} = 2.0\ F_y$	1. Normal loads 2. SRV_{ALL} 3. SSE 4. LOCA	Axial + bending	53,850	23,144

NOTE: Operability assurance is demonstrated by analysis.

Fuel racks have also been evaluated for the appropriate hydrodynamic loads.

TABLE 3.9-2 (t)
REACTOR VESSEL SUPPORT EQUIPMENT
CRD HOUSING SUPPORT

Criteria	Loading	Location	Allowable Stress (psi)	Calculated Stress (psi)
Primary Stress Limit				
AISC Specification for the design, fabrication and erection of structural steel for buildings.				
For Normal & Upset Condition: $f_a = 0.60\ f_y$ (tension) $f_b = 0.66\ f_y$ (bending) $f_v = 0.40\ f_y$ (shear) f_y = Material Yield Strength	(See Note Below)			
For Faulted Condition: F_a limit = 1.5 f_a (tension)	For Faulted Condition Loads:	Beams (Top Chord)	F_a = 33,000	F_a = 14,000
	1. Dead Weight		F_b = 33,000	F_b = 20,000
F_b limit = 1.5 f_b (bending)	2. Impact Force From Failure of a CRD Housing.	Beams (Bottom Chord)	F_a = 33,000	F_a = 11,700
F_v limit = 1.5 f_v (shear)	(Deadweights and earth quake loads are very small compared to impact force.	Grid structure	F_b = 33,000	F_b = 21,000
f_y = Material Yield Strength			F_b = 41,500	F_b = 40,500
			F_v = 27,500	F_v = 11,600

NOTE: Normal and upset, and emergency conditions are not evaluated for this equipment because the housing support only acts to take loads during a specific faulted condition (i.e., the case of a ruptured housing).

TABLE 3.9-2(u)
CONTROL ROD DRIVE

MAIN FLANGE (2) CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
Allowable Primary Membrane Stress plus Bending Stress is based on ASME Boiler & Pressure Vessel Code, Section III for type F304 Stainless Steel @ 575° F S_m=16,675 psi				
For normal and upset condition: $S_{allow} = 1.5 \times S_m$	For normal & upset condition: 1. Normal Loads [1] 2. Scram with OBE 3. Scram with no buffer	General Membrane + Bending	25,000	5,813
For emergency condition: $S_{allow} = 1.8 \times S_m$	For emergency condition: 1. Normal Loads [1] 2. Scram at emergency vessel pressure condition 3. Scram with accumulator at over-pressure	General Membrane + Bending	30,000	4,300
For faulted condition: $S_{allow} = 3.6 \times S_m$	For faulted condition: 1. Normal Loads [1] 2. Scram with SSE 3. Scram with stuck rod	General Membrane + Bending	60,000	7,294

TABLE 3.9-2 (u) (Cont'd)

MAIN FLANGE (2) CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
Allowable Primary Membrane Stress plus Bending Stress is based on ASME Boiler & Pressure Vessel Code, Section III for type F304 Stainless Steel @ 250° F S_m = 20,000 psi				
For normal and upset condition: $S_{allow} = 1.5 \times S_m$	For normal & upset condition: 1. Normal Loads [(1)] 2. Scram with OBE and no buffer	General Membrane + Bending	30,000	17,922
For emergency condition: $S_{allow} = 1.8 \times S_m$	For emergency condition: 1. Normal Loads [(1)] 2. Scram with accumulator at over-pressure	General Membrane + Bending	36,000	1,838
For faulted condition: $S_{allow} = 3.6 \times S_m$	For faulted condition: 1. Normal Loads [(1)] 2. Scram with SSE 3. Scram with stuck rod	General Membrane + Bending	72,000	4,041

TABLE 3.9-2 (u) (Cont'd)

INDICATOR TUBE CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
Allowable Primary Membrane Stress plus Bending Stress is based on ASME Boiler & Pressure Vessel Code, Section III for type 316 Stainless Steel @ 250° F S_m= 19,200				
For normal and upset condition: $S_{allow} = 1.5 \times S_m$	For normal & upset condition: 1. Normal Loads (1) 2. Scram with OBE and no buffer 3. SRV	General Membrane + Bending	28,800	23,700
For emergency condition:	For emergency condition: Normal load upset condition is more severe			
For faulted condition: $S_{allow} = 2.4 \times S_m$	For faulted condition: 1. Normal Loads (1) 2. Scram with SSE 3. Scram with stuck rod	General Membrane	46,100	28,900
Fatique usage = 0.665				

Notes: Normal loads include pressure, temperature, weight and mechanical loads.
Hydrodynamic loads do not significantly contribute to the flange stresses.

TABLE 3.9-2 (v)
CONTROL ROD DRIVE HOUSING

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
Primary Stress Limit – The allowable primary membrane stress is based on the ASME Boiler and Pressure Vessel Code, Section III, for Class I vessels, for type 304 stainless steel.				
For normal and upset condition: S_{limit} = 1.0 x S_m = 16,600 psi @ 575° F	Normal and upset condition loads: 1. Design pressure 2. Stuck rod scram loads 3. Operational basis earthquake, with housing lateral support installed	Maximum membrane stress intensity occurs at the tube to the tube weld near the center of housing for normal, upset and emergency conditions	16,600	15,844
	4. SRV	Membrane + bending	24,900	15,844
For faulted condition: S_{limit} = 2.4 x S_m	Faulted condition: 1. Design pressure 2. Stuck rod scram loads 3. Safe shutdown earthquake, with housing lateral support installed 4. SRV 5. LOCA		39,840	16,400

Note: Emergency condition loads are lower than loads for normal and upset conditions.

TABLE 3.9-2 (w)
JET PUMPS

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
ASME B&PV Code Section III for Type 304 Stainless Steel S_m = 16,867 psi				
For Service Levels A and B (Normal and upset) Conditions $S_{limit} = 3.0\ S_m$	1. Normal 2. OBE 3. SRV 4. Pressure	Primary membrane + Bending + Secondary Membrane	50,700	21,100
For Service Level C (Emergency) Condition $S_{limit} = 2.25\ S_m$	1. Normal 2. OBE 3. SRV 4. Pressure	Primary membrane + Bending	38,025	13,000
For Service Level D (Faulted) Condition $S_{limit} = 3.6\ S_m$	1. Normal 2. SSE 3. LOCA 4. SRV 5. Pressure	Primary membrane + Bending	60,840	60,298

Fatigue range = 0.976

TABLE 3.9-2 (x)
LPCI COUPLING (STRUT-TO-SHROUD WELD)

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
ASME B&PV Code for Type 316L Stainless Steel				
For Service Levels A & B (Normal and upset) conditions $S_{limit} = 1.5 \times 0.7 \times S_m$	1. Normal 2. Pressure 3. OBE 4. SRV	Primary membrane + Bending	14,860	2,893
For Service Level C (Emergency) condition $S_{limit} = 2.25 \times 0.7 \times S_m$	1. Normal 2. Pressure 3. Chugging 4. SRV	Primary membrane + Bending	22,290	4,795
For Sevice Level D (Faulted) condition $S_{limit} = 3.6 \times 0.7 \times S_m$	1. Normal 2. Pressure 3. LOCA 4. SSE	Primary membrane + Bending	35,658	22,280

TABLE 3.9-2(y)
CONTROL ROD GUIDE TUBE

CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
CONTROL ROD GUIDE TUBE				
Primary Stress limit				
The allowable primary membrane stress plus bending stress is based on the ASME Boiler and Pressure Vessel Code, Section III, Class for Type 304 Stainless Steel material				
For Service Levels A & B (normal and upset) [(1)] conditions: $1.5\ S_m = 1.5 \times 16{,}000$ $= 24{,}000$ psi	1. External pressure 2. Metal and water weight 3. OBE 4. SRV 5. Scram	Primary membrane + Bending	24,000	10,097
For Service Level D (faulted) condition $S_{limit} = 2.4\ S_m$ $= 2.4 \times 16{,}000$ $= 38{,}400$ psi	1. External pressure 2. Metal plus water weight 3. OBE 4. SSE 5. Scram	Primary membrane + Bending	38,400	17,086

Note: (1) Normal, upset and emergency loads are analyzed together and compared to normal/upset allowables.

TABLE 3.9-2 (z)
INCORE HOUSING

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Primary Stress Limit – The allowable primary membrane stress is based on ASME Boiler and Pressure Vessel Code, Section III for Class 1 vessels for type Inconel 600 austenitic high nickel alloy steel				
For Service Levels A and B (normal and upset conditions) S_{limit} = 1.0 S_m = 23,300 psi at 575× F	Service Levels A and B (Normal and Upset Condition) Loads 1. Design pressure 2. OBE 3. SRV	Maximum membrane stress intensity occurs at the outer surface of the vessel penetration	23,300	18,055
Service Levels D (Faulted Condition) Stress limit is the lesser of 0.7 Su = 0.7 (5.24) = 50,000 or 2.4Sm = 24 (23,300) = 55,920	Service Level D (Faulted Condition) Loads 1. Design Pressure 2. Static Weights 3. Safe Shutdown Earthquake 4. LOCA		50,000	22,655

TABLE 3.9-2(aa)
HIGH PRESSURE CORE SPRAY SYSTEM ASME CODE CLASS 1 VALVE

	COMPONENT/LOAD/STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE LIMIT	DESIGN/ CALCULATED VALUE
1.	Body and Bonnet			
1.1	Loads:			
	Design Pressure	System requirement	1,575 psi	
	Design Temperature	System requirement	575° F	
	Pipe Reaction	Not specified		
	Thermal Effects	Not specified		
1.2	Pressure Rating, psi	ASME Section III [1] Figure NB 3545.1-2	Pr = 655 psi	Pr = 655 psi
1.3	Minimum Wall Thickness, inches	ASME Section III Para NB-3542	t (nominal) = 1.125 inches	t_n = 0.94 min inches
1.4	Primary Membrane Stress, psi	ASME Section III Para NB-3545-1	$P_n \leq S_n$ (500° F) = 19,400 psi	P_n = 7,256 psi
1.5	Secondary Stress Due to Pipe Reaction	ASME Section III Para NB-3545.2. (b) (S = 30,000 psi)	Pe = Greatest value of Ped, Peb and Pet $\leq 1.5\ S_m$ (500° F) 1.5 (19,400) = 29,100 psi	Ped = 6,919 psi Peb = 13,623 psi Pet = 13,623 psi Pe = Pet = 13,623 psi

[1] ASME Code Section III, 1971 Edition

TABLE 3.9-2 (aa) (Cont'd)

	COMPONENT/LOAD/STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE LIMIT	DESIGN/ CALCULATED VALUE
1.6	Primary plus secondary stress due to internal pressure	ASME Section III Para NB-3545.2 (a)		Qp = 20,138 psi
1.7	Thermal secondary stress	ASME Section III Para NB 3545.2 (c)		Qt = 598 psi
1.8	Sum of primary plus secondary stress	ASME Section III Para NB-3545.2		S_n = Qp+Pe+2QT = 27,253 psi
1.9	Fatigue requirements	ASME Section III Para NB-3545.3	NA ≥ 2010 cycles	N_a = 70,000 cycles
1.10	Cyclic rating	ASME Section III Para NB-3550	It ≤ 1.0	It = 0.38

TABLE 3.9-2 (aa) (Cont'd)

COMPONENT/LOAD/STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE LIMIT	DESIGN/ CALCULATED VALUE
2.0 <u>Body to bonnet bolting</u>			
2.1 Loads: design pressure and temperature, gasket loads, stem operational load, seismic load (design basis earthquake)	ASME Section III Para NB-3647.1	--	--
2.2 Bolt area	ASME Section III (1) Para NB 3647.1	$A_b \geq 17.64\ in^2$	$A_b = 18.9\ in^2$
2.3 <u>Body flange stresses</u>	ASME Section III Para NB-3647.1	--	--
2.3.1 (1) Operating conditions		$S_h \leq 1.5\ S_m$ (500× F) =29,100 psi $S_r \leq 1.5\ S_m$ =29,100 psi $S_t \leq 1.5\ S_m$ =29,100 psi	S_h = 25,400 psi S_r = 22,500 psi S_t = 11,300 psi

TABLE 3.9-2 (aa) (Cont'd)

COMPONENT/LOAD/STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE LIMIT	DESIGN/ CALCULATED VALUE
2.3.2 (II) Gasket seating Condition	ASME Section III Para NB-3647.1	$S_h \leq 1.5\ S_m$ (500° F) =29,100 psi $S_R \leq 1.5\ S_m$ =29,100 psi $S_T \leq 1.5\ S_m$ =29,100 psi	S_h = 23,400 psi S_R = 20,300 psi S_T = 10,200 psi

TABLE 3.9-2 (aa) (Cont'd)

COMPONENT/LOAD/STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE LIMIT	DESIGN/ CALCULATED VALUE
3.1 Buckling on stem	Calculate Slenderness Ratio. If greater than 30, calculate allowable load from Rankine's Formula using safety factor of 4	Slenderness Ratio $\leq$ 30	Slenderness ratio = 28.2 Therefore, no buckling
4.0 <u>Disc Analysis</u>	--	--	--
4.1 Loads: Maximum Differential Pressure (2)			
4.2 Maximum Stress in the Disc	ASME Section III Para NB-3215 and ASME Section III Para NB-3221.3	$S_{max} \leq 1.5\ S_m$ (500° F) = 29,100 psi	Max Stress = 11,530 psi

TABLE 3.9-2 (aa) (Cont'd)

COMPONENT/LOAD/STRESS TYPE	DESIGN PROCEDURE	ALLOWABLE LIMIT	DESIGN/ CALCULATED VALUE
5.0 <u>Yoke and Yoke Connections</u>			
5.1 Loads: Stem Operational Load	Calculate Stresses in the yoke and yoke connections to acceptable structural analysis methods.	--	--
5.2 Tensile stress in Yoke Leg Bolts		S_{max} £ S_m (100° F) = 103,000 psi	S_{max} = 49,000 psi

TABLE 3.9-3

(THIS TABLE HAS BEEN DELETED)

Table 3.9-4
NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48

		NRC Regulatory Guide 1.48			Plant Name			
Component	Plant Condition	Loading Combination 1/	Design Limit	Regulatory Guide Paragraph	Loading Combination (f)	Code Allowable Stresses	ASME Section III Reference	Comparison with NRC Regulatory Guide 1.48
Class 1 Vessels	Upset (U) Emergency (E) Faulted (F)	[NPC or UPC] + 0.5 SSE (1) EPC NPC + SSE + DSL	NB – 3323 NB – 3224 2/ NB – 3225	1.a 1.b 1.c	[NPC or UPC] + 0.5 SSE (1) EPC NPC + SSE + DSL	1.08m (includes secondary stresses) 1.8Sm or 1.5Sy App.F - Sect III	NB-3223 NB-3224 NB-3225	Agree
Class 1 Piping	U E F	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	NB – 3654 NB – 3655 2/ NB – 3656	1.a 1.b 1.c	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	3.0Sm (includes secondary stresses) 2.25Sm 3.0Sm	NB-3654 NB-3655 NB-3656	Agree
Class 1 Pumps (Inactive)	U E F	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	NB – 3223 5/ NB – 3224 1/ NB – 3225	2.a 2.b 2.c	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	1.65Sm (excludes secondary stresses) 1.8Sm App. F-Section III	NB-3223 NB-3224	Agree
Class 1 Pumps (Active)	U E F	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	NB-3222 5/ NB-3222 6/ NB-3222 7/ 8	4.a 4.a 4.a	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	Not Applicable	Not Applicable	Not Applicable
Class 1 Valves (Inactive) by analysis	U E F	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	NB – 3223 5/ ND – 3224 4/ NB – 3225 2/	2.a 2.b 2.c	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	Not Applicable	Not Applicable	Not Applicable
Class 1 Valves (Inactive) designed by either Std. of alternative design rules	U E F	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	1.1 Pr 1.2 Pr 1.5 Pr	3.a 3.b 3.c	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	1.1 Pr 1.2 Pr 1.5 Pr	NB-3525 NB-3526 NB-3527	Agree

Table 3.9-4
NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48

Component	Plant Condition	NRC Regulatory Guide 1.48			Plant Name			Comparison with NRC Regulatory Guide 1.48
		Loading Combination 1/	Design Limit	Regulatory Guide Paragraph	Loading Combination (f)	Code Allowable Stresses	ASME Section III Reference	
Class 1 Valves (Active) by Analysis	U E F	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	NB – 3222 5/ NB – 3222 6/ NB – 3222 7/ 8/	4.a 4.b 4.c	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	Not Applicable	Not Applicable	Not Applicable
Class 1 Valves (Active) designed by Std. or alternative design rules	U E F	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	1.0Pr 1.0Pr 6/ 1.0Pr	5.a 5.b 5.c	[NPC or UPC] + 0.1 SSE EPC NPC + SSE + DSL	1.0Pr 1.0Pr (a) 1.0Pr	NB-3525 NB-3526 NB-3527	Agree
Class 2 & 3 Vessels (Division 1) of Section of the ASME Code	U E F	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	1.1S 1.1S 9/ 1.5S	6.a 6.b 6.c	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	$\sigma m = 1.1S$ (c) $\sigma m = 2.0S$	code case 1607, NC/ND-3300	Agree except for Faulted Condi-tion NRC more conservative
Class 2 Vessels (Division 2) of Section VIII of the ASME Code	U E F	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	NB – 3223 NB – 3224 2/ NB – 3225	7.a 7.b 7.c	[NPC or UPC] + 0.5 SS EPC NPC + SSE + DSL	Not Applicable	Not Applicable	Not Applicable
Class 2 & 3 Piping	U E F	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	NC – 3611.1(b) (4)(c)(b)(1) NC – 3611.1(b) (4)(c)(b)(1) 10/ NC – 3611.1(b) (4)(c)(b)(2)	8.a 8.a 8.b	[NPC or UPC] + 0.5 SSE EPC NPC + SSE + DSL	1.2 Sh 1.8 Sh 2.4 Sh	NC/ND-3611.3(b) NC/ND-3611.3(c)	NRC more conservative GE reflects industry position

Table 3.9-4
NSSS COMPLIANCE WITH REGULATORY GUIDE 1.48

Component	Plant Condition	NRC Regulatory Guide 1.48: Loading Combination 1/	NRC Regulatory Guide 1.48: Design Limit	NRC Regulatory Guide 1.48: Regulatory Guide Paragraph	Plant Name: Loading Combination(f)	Plant Name: Code Allowable Stresses	Plant Name: ASME Section III Reference	Comparison with NRC Regulatory Guide 1.48
Class 2 & 3 Pumps (Inactive)	U	[NPC or UPC] + 0.5 SSE	$\sigma_m \leq 1.1\,S \geq \frac{\sigma_m + \sigma_b}{1.5}$	9.a	[NPC or UPC] + 0.5 SSE	Not Applicable	Not Applicable	Not Applicable
	E	EPC	$\sigma_m \leq 1.1\,S \geq \frac{\sigma_m + \sigma_b}{1.5}$	9.a	EPC			
	F	NPC + SSE + DSL	$\sigma_m \leq 1.1\,S \geq \sigma_m + \sigma_b$	9.b	NPC + SSE + DSL			
Class 2 & 3 Pumps (Active)	U	[NPC or UPC] + 0.5 SSE	$\sigma_m \leq 1.0\,S \geq \frac{\sigma_m + \sigma_b}{1.5}$	10.a	[NPC or UPC] + 0.5 SSE	$\left.\begin{array}{l}\sigma_m = 1.1S \\ \sigma_m = 1.25\end{array}\right\}$ (a) (c)	Code case 1636, NC/ND-3423 (see Note (b))	Agree (a)
	E	EPC	$\sigma_m \leq 1.0\,S \geq \frac{\sigma_m + \sigma_b}{1.5}$	10.a	EPC			
	F	NPC + SSE + DSL	$\sigma_m \leq 1.0\,S \geq \frac{\sigma_m + \sigma_b}{1.5}$ 11/	10.a	NPC + SSE + DSL			
Class 2 & 3 Valves (Inactive)	U	[NPC or UPC] or 0.5 SSE	1.1 Pr	11.a	[NPC or UPC] + 0.5 SSE	$\left.\begin{array}{l}\sigma_m = 1.1S \\ \sigma_m = 2.0S\end{array}\right\}$ (c)	code case 1635, NC/ND-3521 (see Note (1))	Equally conservative
	E	EPC	1.1 Pr	11.a	EPC			
	F	NPC + SSE + DSL	1.2 Pr	11.b	NPC + SSE + DSL			
Class 2 & 3 Valves (Active)	U	[NPC or UPC] + 0.5 SSE	$\left.\begin{array}{l}1.0\,Pr \\ 1.0\,pr \\ 1.0\,pr\end{array}\right\}$ 11/	12.a	[NPC or UPC] + 0.5 SSE	$\left.\begin{array}{l}\sigma_m = 1.1S \\ \sigma_m = 1.2S\end{array}\right\}$ (a) (c)	code case 1635 NC/ND-3521 (see Note (b))	Equally conservative (e)
	E	EPC		12.a	EPC			
	F	NPC + SSE + DSL		12.a	NPC + SSE + DSL			

TABLE 3.9-4
NOTES FOR COMPARISON TABLE 3.9-4

Numerical indicators in the regulatory guide portion of the table correspond to footnotes of the Regulatory Guide 1.48.

Alphabetical indicators in the table (or comparative column) correspond to the following:

a. In addition to compliance with the design limits specified, assurance of operability under all design loading combinations shall be in accordance with subsection 3.9.3.2.

b. Not used

c. The design limit for local membrane stress intensity or primary membrane plus primary bending stress intensity is 150 percent of that allowed for general membrane (except as limited to 2.4S for inactive components under faulted condition).

d. Not used

e. Inactive limits may be used since operability will be demonstrated in accordance with Section 3.9.3.2.

f. When selecting plant events for evaluation, the choice of the events to be included in each plant condition is selected based on the probability of occurrence of the particular load combination. The combination of loads are those identified in Table.

TABLE 3.9-5
BOP ACTIVE VALVES AND PUMPS

SYSTEM	VALVES	SYSTEM	VALVES
Feedwater	1B21-F010A	Main Steam	1B21-F037S
	1B21-F010B	(Cont'd)	1B21-F039B
	1B21-F032A		1B21-F039C
	1B21-F032B		1B21-F039D
	1B21-F065A		1B21-F039E
	1B21-F065B		1B21-F039H
	1B21-F433A		1B21-F039K
	1B21-F433B		1B21-F039S
Main Steam	1B21-F016		1B21-F067A
	1B21-F019		1B21-F067B
	1B21-F024A		1B21-F067C
	1B21-F024B		1B21-F067D
	1B21-F024C		1B21-F078A
	1B21-F024D		1B21-F078B
	1B21-F029A		1B21-F078C
	1B21-F029B		1B21-F078D
	1B21-F029C		1B21-F078E
	1B21-F029D		1B21-F078F
	1B21-F036A		1B21-F078G
	1B21-F036F		1B21-F078H
	1B21-F036G		1B21-F078J
	1B21-F036J		1B21-F078K
	1B21-F036L		1B21-F078L
	1B21-F036M		1B21-F078M
	1B21-F036N		1B21-F078N
	1B21-F036P		1B21-F078P
	1B21-F036R		1B21-F078R
	1B21-F037A		1B21-F078S
	1B21-F037B		1B21-F379A
	1B21-F037C		1B21-F379B
	1B21-F037D		1B21-F379C
	1B21-F037E		1B21-F379D
	1B21-F037F		1B21-F379E
	1B21-F037G		1B21-F379F
	1B21-F037H		1B21-F379G
	1B21-F037J		1B21-F379H
	1B21-F037K		1B21-F379J
	1B21-F037L		1B21-F379K
	1B21-F037M		1B21-F379L
	1B21-F037N		1B21-F379M
	1B21-F037P		1B21-F379N
	1B21-F037R		1B21-F379P
			1B21-F379Q
			1B21-F379R

TABLE 3.9-5 (Cont'd)
BOP ACTIVE VALVES AND PUMPS

SYSTEM	VALVES	SYSTEM	VALVES
Reactor	1B33-F019	Residual Heat	1E12-F031C
Recirculation	1B33-F020	Removal (Cont'd)	1E12-F036
			1E12-F037A (Note 2)
Control Rod	1C11-F083		1E12-F037B (Note 2)
Drive	1C11-F122		1E12-F040
			1E12-F041A
Standby Liquid	1C41-F001A		1E12-F041C
Control	1C41-F001B		1E12-F041B
	1C41-F006		1E12-F042A
	1C41-F029A		1E12-F042B
	1C41-F029B		1E12-F042C
	1C41-F033A		1E12-F046A
	1C41-F033B		1E12-F046B
	1C41-F336		1E12-F046C
			1E12-F047A
Residual Heat	1E12-F003A		1E12-F047B
Removal	1E12-F003B		1E12-F048A
	1E12-F004A		1E12-F048B
	1E12-F004B		1E12-F049
	1E12-F005		1E12-F050A
			1E12-F050B
			1E12-F053A
	1E12-F008		1E12-F053B
	1E12-F009		1E12-F054A
	1E12-F014A		1E12-F054B
	1E12-F014B		1E12-F060A
	1E12-F017A		1E12-F060B
	1E12-F017B		1E12-F064A
	1E12-F019		1E12-F064B
	1E12-F021		1E12-F064C
	1E12-F023		1E12-F068A
	1E12-F024A		1E12-F068B
	1E12-F024B		1E12-F112A
	1E12-F025A		1E12-F112B
	1E12-F025B		
	1E12-F025C		
	1E12-F027A		
	1E12-F027B		
	1E12-F028A		
	1E12-F028B		
	1E12-F031A		
	1E12-F031B		

TABLE 3.9-5 (Cont'd)
BOP ACTIVE VALVES AND PUMPS

SYSTEM	VALVES	SYSTEM	VALVES
Residual Heat		High Pressure	1E22-F024
Removal (Cont'd)	1E12-F075A	Core Spray (Cont'd)	1E22-F035
	1E12-F075B		1E22-F039
	1E12-F084A		1E22-F330
	1E12-F084B		1E22-F332
	1E12-F084C		
	1E12-F085A	Leak Detection	1E31-F014
	1E12-F085B		1E31-F015
	1E12-F085C		1E31-F017
	1E12-F094 (Note 2)		1E31-F018
	1E12-F096		
	1E12-F098		
	1E12-F101	MSIV Leakage	
	1E12-F105	Control	
	1E12-F496		
	1E12-F497		
	1E21-F001		
	1E21-F003		
	1E21-F005		
Low Pressure	1E21-F006		
Core Spray	1E21-F011		
	1E21-F012		
	1E21-F018		
	1E21-F031		
	1E21-F033		
	1E21-F034		
	1E21-F303		
		Reactor Core	1E51-F010
	1E22-F002	Isolation Cooling	1E51-F011
	1E22-F005		1E51-F013
High Pressure	1E22-F006		1E51-F015
Core Spray	1E22-F007		1E51-F018
	1E22-F014		1E51-F019
	1E22-F016		1E51-F021
			1E51-F022
			1E51-F025

TABLE 3.9-5 (Cont'd)
BOP ACTIVE VALVES AND PUMPS

SYSTEM	VALVES	SYSTEM	VALVES
Reactor Core	1E51-F026	Component Cooling	1CC075A
Isolation	1E51-F030	(Cont'd)	1CC075B
Cooling (Cont'd)	1E51-F031		1CC076A
	1E51-F040		1CC076B
	1E51-F045		1CC127
	1E51-F046 (Note 1)		1CC128
	1E51-F059		1CC280A
	1E51-F061		1CC280B
	1E51-F062		
	1E51-F063		1CM002A
	1E51-F064		1CM002B
	1E51-F065		1CM003A
	1E51-F066		1CM003B
	1E51-F068 (Note 3)	Containment	1CM011
	1E51-F076	Monitoring	1CM012
	1E51-F077		1CM014
	1E51-F078		1CM015
	1E51-F079		
	1E51-F081		1CM017
	1E51-F090		1CM018
	1E51-F377A		1CM022
	1E51-F377B		1CM023
	1E51-F004		1CM025
	1E51-F005		1CM026
			1CM028
	1G33-F001		1CM031
Reactor Water	1G33-F004		1CM032
Cleanup	1G33-F028		
	1G33-F034		1CM034
	1G33-F039		1CM047
	1G33-F040		1CM048
	1G33-F051		1CM051
	1G33-F052A		1CM053
	1G33-F052B		1CM066
	1G33-F053		1CM067
	1G33-F054		
		Cycled	1CY016
	1CC049	Condensate	1CY017
Component Cooling	1CC050		
	1CC053		
	1CC054		
	1CC057		
	1CC060		

TABLE 3.9-5 (Cont’d)
BOP ACTIVE VALVES AND PUMPS

SYSTEM	VALVES	SYSTEM	VALVES
Fuel Pool	1FC004A	Instrument Air	1IA008
Cooling and	1FC004B	(Cont'd)	1IA012A
Cleanup	1FC007		1IA012B
	1FC008		1IA013A
	1FC011A		1IA013B
	1FC011B		1IA042A
	1FC013A		1IA042B
	1FC013B		1IA044A
	1FC015A		1IA044B
	1FC015B		1IA128A
	1FC016A		1IA128B
	1FC016B		1IA175
	1FC017		
	1FC023	Makeup	0MC009
	1FC024A	Condensate	0MC010
	1FC024B		
	1FC026A	Breathing Air	0RA026
	1FC026B		0RA027
	1FC036		0RA028
	1FC037		0RA029
	1FC091		1RA016A
			1RA016B
Fire Protection	1FP050		1RA022A
	1FP052		1RA022B
	1FP053		1RA023A
	1FP092		1RA023B
		Process Sampling	1PS004
			1PS005
Containment	1HG001		1PS009
Combustible	1HG004		1PS010
Gas Control	1HG005		1PS016
	1HG008		1PS017
	1HG009A		1PS022
	1HG009B		1PS023
	1HG010A		1PS031
	1HG010B		1PS032
	1HG010C		1PS034
	1HG010D		1PS035
	1HG011A		
	1HG011B		
	1HG011C		1PS043A
	1HG011D		1PS043B
			1PS044A
Instrument Air	1IA005		1PS044B
	1IA006		
	IIA007		
			1PS055

TABLE 3.9-5 (Cont'd)
BOP ACTIVE VALVES AND PUMPS

SYSTEM	VALVES
Process Sampling (Cont'd)	1PS056
	1PS069
	1PS070
Reactor Building Equipment Drain	1RE019
	1RE020
	1RE021
	1RE022
Containment Building Floor Drain	1RF019
	1RF020
	1RF021
	1RF022
Service Air	1SA029
	1SA030
	1SA031
	1SA032
Suppression Pool Cleanup	1SF001
	1SF002
	1SF004
Suppression Pool Makeup	1SM001A
	1SM001B
	1SM002A
	1SM002B
	1SM003A
	1SM003B
	1SM008
	1SM009
	1SM010
	1SM011
Shutdown Service Water	1SX001A
	1SX001B
	1SX001C
	1SX006C
	1SX008A

SYSTEM	VALVES
Shutdown Service Water (Cont'd)	1SX008B
	1SX008C
	1SX010A
	1SX010B
	1SX010C
	1SX012A
	1SX012B
	1SX013D
	1SX013E
	1SX013F
	1SX014A
	1SX014B
	1SX014C
	1SX016A
	1SX016B
	1SX019A
	1SX019B
	1SX020A
	1SX020B
	1SX023A
	1SX023B
	1SX025A
	1SX025B
	1SX025C
	1SX027A
	1SX027B
	1SX027C
	1SX029A
	1SX029B
	1SX029C
	1SX033
	1SX037
	1SX041A
	1SX041B
	1SX062A
	1SX062B
	1SX063A
	1SX063B

TABLE 3.9-5 (Cont'd)
BOP ACTIVE VALVES AND PUMPS

SYSTEM	VALVES	SYSTEM	VALVES
Shutdown Service		Control Room	0VC010A
Water (Cont'd)		HVAC	0VC010B
			0VC017A
	1SX082A		0VC017B
	1SX082B		0VC020A
			0VC020B
			0VC022A
			0VC022B
			0VC025A
	1SX153A		0VC025B
	1SX153B		
	1SX169C		
	1SX181A		
	1SX181B		
	1SX185A		
	1SX185B		
	1SX189		
	1SX193A		
	1SX193B		
	1SX197		
	1SX208A		
	1SX208B		
	1SX209 (Note 4)		

TABLE 3.9-5 (Cont'd)
BOP ACTIVE VALVES AND PUMPS

SYSTEM	VALVES	SYSTEM	VALVES
Drywell HVAC	1VP004A	Chilled Water	1WO552A
	1VP004B	(Cont'd)	1WO552B
	1VP005A		1WO570A
	1VP005B		1WO570B
	1VP014A		
	1VP014B	Radwaste	1WX019
	1VP015A		1WX020
	1VP015B		
	1VP023A	Diesel Oil	1DO001A
	1VP023B		1DO001B
	1VP027A		1DO001C
	1VP027B		1DO005A
			1DO005B
			1DO005C
Drywell Purge	1VQ002	Refrigeration	1RG06MA
	1VQ003		1RG06MB
	1VQ004A		1RG06MC
	1VQ004B		1RG06MD
	1VQ005		1RG06ME
			1RG06MF
			1RG06MG
			1RG06MH
Containment	1VR001A		1RG06MJ
Building HVAC	1VR001B		1RG06MK
			1RG07MA
			1RG07MB
	1VR006A		1RG07MC
	1VR006B		1RG07MD
	1VR007A		1RG07ME
	1VR007B		1RG07MF
	1VR016A		1RG07MG
	1VR016B		1RG07MH
	1VR018A		1RG07MJ
	1VR018B		1RG07MK
	1VR035		1RG12MA
	1VR036		1RG12MB
	1VR040		1RG12MC
	1VR041		
		Standby Gas	1VG056B
Chilled Water	1WO001A	Treatment	1VG057B
	1WO001B		
	1WO002A		
	1WO002B		
	1WO551A		
	1WO551B		

TABLE 3.9-5 (Cont'd)
BOP ACTIVE VALVES AND PUMPS

SYSTEM	PUMPS
Diesel Oil	1DO01PA
	1DO01PB
	1DO01PC
Control Room HVAC	0VC08PA
	0VC08PB
RHR Water Leg	1E12-C003
LPCS Water Leg	1E21-C002
HPCS Water Leg	1E22-C003
RCIC	1E51-C003
Shutdown Service Water	1SX01PA
	1SX01PB
	1SX01PC
Fuel Pool Cooling & Cleanup	1FC02PA
	1FC02PB

SYSTEM	COMPONENT
Diesel Generator	1DG01KA
	1DG01KB
	1DG168
	1DG169
	1DG170
	1DG171
	1DG172
	1DG173
	1DG006A
	1DG006B
	1DG006C
	1DG006D
	1DG006E
	1DG006F
	1DG008A
	1DG008B
	1DG008C
	1DG008D
	1DG008E
	1DG008F
	1DG008G
	1DG008H
	1DG008J
	1DG008K

TABLE 3.9-5 (Cont'd)
BOP ACTIVE VALVES AND PUMPS
NOTES

1. Valve adminstatively controlled in the open position with supply breaker open.

2. Valve is closed, and the supply breaker opened during normal ops to mitigate potential adverse effects of a fire caused short. The Out-of-Service alarm is defeated on the MCB.

3. To prevent spurious operation as a result of a fire induced fault, 1E51-F068 is maintained in the open position with the shorting switch in the shorting position.

4. Valve 1SX209 has been abandoned in place and deactivated in the open position.

TABLE 3.9-6
NSSS SEISMIC ACTIVE PUMPS AND VALVES

COMPONENT NAME	IDENTIFICATION AS SHOWN ON APPLICABLE FIGURES
Control Rod Drive Valves	1C11-D001/114
	1C11-D001/115
	1C11-D001/126
	1C11-D001/127
	1C11-D001/138
	1C11-D001/139
	1C11-F009
	1C11-F010
	1C11-F011
	1C11-F180
	1C11-F181
	1C11-F182
High Pressure Core Spray Pumps	1E22-C001
High Pressure Core Spray Valves	1E22-F001
	1E22-F004
	1E22-F010 (Note 1)
	1E22-F011
	1E22-F012
	1E22-F015
	1E22-F023
Low Pressure Core Spray Pump	1E21-C001
Main Steam Valves	1B21-F022A
	1B21-F022B
	1B21-F022C
	1B21-F022D
	1B21-F028A
	1B21-F028B
	1B21-F028C
	1B21-F028D
	1B21-F041A
	1B21-F041B
	1B21-F041C
	1B21-F041D

Note 1 – Valve is closed and the breakers opened during normal operation in response to the NRC Hot Short concerns. The Out-of-Service alarm is defeated from the MCB.

TABLE 3.9-6 (Cont'd)

COMPONENT NAME	IDENTIFICATION AS SHOWN ON APPLICABLE FIGURES
Main Steam Valves (Cont'd)	1B21-F041F 1B21-F041G 1B21-F041L 1B21-F047A 1B21-F047B 1B21-F047C 1B21-F047D 1B21-F047F 1B21-F051B 1B21-F051C 1B21-F051D 1B21-F051G
Reactor Core Isolation Cooling Pump	1E51-C001
Reactor Core Isolation Cooling Turbine	1E51-C002
Reactor Core Isolation Cooling Valve	1E51-C002E
Residual Heat Removal Pumps	1E12-C002A 1E12-C002B 1E12-C002C
Standby Liquid Control Pumps	1C41-C001A 1C41-C001B
Standby Liquid Control Valves	1C41-F004A 1C41-F004B
Diesel Generator	1DGOIKC

Tables 3.9-7 through 3.9-12 have been deleted intentionally.

TABLE 3.9-13
NON-NSSS PIPING TO BE TESTED FOR THERMAL EXPANSION
OPERATING AND TRANSIENT VIBRATION

SYSTEM	FROM	TO	TRANSIENTS	PARAMETERS MONITORED	METHOD OF MONITORING DISPLACEMENT
PRE-OPERATIONAL THERMAL EXPANSION TEST					
RHR Shutdown Cooling	Recir. Lines	Heat Exchangers	--	Temperatures Thermal Expansion	Visual
PRE-OPERATIONAL STEADY STATE VIBRATION TEST					
HPCS	RCIC Storage/ Suction Strainer	RPV	--	Steady State Vibration	Visual
LPCS Suction	RPV Strainer		--	Steady State Vibration	Visual
RHR					
LPCI Suction	RPV Strainer		--	Steady State Vibration	Visual
Cont. Spray Suction Strainer	Bypass Piping		--	Steady State Vibration	Visual
Suppression Pool Cooling	Suction Strainer	Heat Exchangers	--	Steady State Vibration	Visual

TABLE 3.9-13 (Cont'd)

SYSTEM	FROM	TO	TRANSIENTS	PARAMETERS MONITORED	METHOD OF MONITORING DISPLACEMENT
Shutdown Cooling	Recirc. Lines	Heat Exchangers	--	Steady State Vibration	Visual
STANDBY LIQUID CONTROL	Pumps	RPV	--	Steady State Vibration	Visual
ESSENTIAL INSTRUMENTATION					
RPV Level Sensing	RPV	Drywell Wall	--	Steady State Vibration	Visual
CRD Insert and Withdrawal	RPV Wall	Drywell	--	Steady State Vibration	Visual
PRE-OPERATIONAL TRANSIENT VIBRATION TEST					
FEEDWATER	Motor Driven Feedwater Pump	Condenser	FWP Trip	Dynamic Disp.	Visual
HPCS	RCIC Storage/ Suction Strainer	RPV	Pump Starts Pump Trips	Dynamic Disp.	Visual
LPCS	Suction Strainer	RPV	Pump Starts Pump Trips	Dynamic Disp.	Visual

TABLE 3.9-13 (Cont'd)

SYSTEM	FROM	TO	TRANSIENTS	PARAMETERS MONITORED	METHOD OF MONITORING DISPLACEMENT
RHR					
LPCI	Suction Strainer	RPV	Pump Starts Pump Trips	Dynamic Disp.	Visual
Cont.	Suction Spray	Bypass Strainer	Pump Starts Piping	Dynamic Disp.	Visual
Suppression Pool Cooling	Suction Strainer	Heat Exchangers	Pump Starts Pump Trips	Dynamic Disp.	Visual
Shutdown Cooling	Recirc. Lines	Heat Exchangers	Pump Starts Pump Trips	Dynamic Disp.	Visual
STANDBY LIQUID CONTROL	Pumps	RPV	Pump Starts Pump Trips	Dynamic Disp.	Visual
ESSENTIAL INSTRUMENTATION					
CRD Insert and Withdrawal	RPV	Drywell Wall	Scram	Dynamic Disp.	Visual
START-UP THERMAL EXPANSION TEST					
MAIN STEAM Main Steam Lines	RPV	Turbine Stop Valves	--	Temperatures Thermal Disp.	Instrumented

TABLE 3.9-13 (Cont'd)

SYSTEM	FROM	TO	TRANSIENTS	PARAMETERS MONITORED	METHOD OF MONITORING DISPLACEMENT
START-UP THERMAL EXPANSION TEST (Con't)					
SRV Dischg. Piping	SRV	Drywell Penetration	--	Temperatures Thermal Disp.	Visual * inspection and/or Instrumented
FEEDWATER					
FW	FWP	RPV	--	Temperatures Thermal Disp.	Visual and/or Instrumented
RCIC	Main Steam	RCIC Turbine	--	Temperature Thermal Disp.	Visual and/or Instrumented
START-UP TRANSIENT VIBRATION TEST					
MAIN STEAM					
Main Steam Lines	MSIV	Turbine Stop Valves	SV Closure	Dynamic Disp.	Instrumented
SRV Dischg.	SRV	Suppression Pool	SRV Discharge	Dynamic Disp.	Visual inspection * and/or Instrumented

TABLE 3.9-13 (Cont'd)

SYSTEM	FROM	TO	TRANSIENTS	PARAMETERS MONITORED	METHOD OF MONITORING DISPLACEMENT
FEEDWATER					
FW	FWP	RPV	FWP Trip	Dynamic Disp.	Instrumented
RCIC	Main Steam	RCIC Turbine	Turbine/Pump Starts	Dynamic Disp.	Visual and/or Instrumented
	Suction/ Storage	RPV	Turbine/Pump Trips	Dynamic Disp.	Visual and/or Instrumented
ESSENTIAL INSTRUMENTATION					
Main Steam Flow for Isolation	Main Steam	Drywell Wall	Steady State Only	Displacement	Visual * Inspection
RCIC Steam Flow for Isolation	RCIC Line	Drywell Wall Instrument Panel	RCIC Pump Operation	Displacement	Visual * Inspection
MAIN STEAM					
Main Steam Lines	Containment Penetration	Turbine Stop Valve	--	Steady State Vibration	Visual * Inspection

TABLE 3.9-13 (Cont'd)

SYSTEM	FROM	TO	TRANSIENTS	PARAMETERS MONITORED	METHOD OF MONITORING DISPLACEMENT
Turbine Bypass Lines	Main Steam	Condenser	--	Steady State Vibration	Visual * Inspection
FEEDWATER	FWP	RPV	--	Steady State Vibration	Visual * Inspection
RCIC	Main Steam	RCIC Turbine	--	Steady State Vibration	Visual * Inspection

* Prior to operations which effect the system, a walkdown inspection will be completed. This is to insure that the system has been constructed in a manner which will preclude excessive vibration. After the operations have been completed, an additional walkdown will be made to inspect for any indication of excessive vibration.

TABLE 3.9-14
STRESS LIMITS FOR DUCTWORK AND DUCT SUPPORTS

	Load Combination	Stress Limits: Duct Supports (Hangers)	Stress Limits: Ductwork	Plant Condition
1.	Normal (N) (Weight + Pressure + Thermal)	AISC Allowable Values	σ_m = 0.6 S_y	Normal
2.	Load Cases 2 & 3 from Table A3.9-6	33% Increase in AISC Allowable Values	σ_m = 0.6 S_y σ_t = 0.95 S_y	Upset
3.	Load Cases 4 through 15 from Table A3.9-6	0.95 S_y	σ_m = 0.95 S_y σ_t = 0.95 S_y	Faulted

KEY

σ_m - Membrane Stress

σ_t - (Membrane + Bending) Stress

S_y - Yield Stress at Corresponding Temperature

2% - Damping values for OBE and pool dynamic loads

4% = Damping values for SSE and pool dynamic loads

TABLE 3.9-15
NSSS Valve/Valve Operator - Seismic Qualification Package
(Q&R MEB (DSER) 76)

VALVE NUMBER	SEISMIC QUALIFICATION PACKAGE*	VALVE NUMBER	SEISMIC QUALIFICATION PACKAGE*
1B33-F023A	SQ-CL732	1E12-F301C	SQ-CL047
1B33-F023B	SQ-CL732	1E21-F340	SQ-CL047
1B33-F060A	SQ-CL730	1E22-F304	SQ-CL047
1B33-F060B	SQ-CL730	1E51-F019	SQ-CL129
1B33-F067A	SQ-CL731	1E41-F076	SQ-CL230
1B33-F067B	SQ-CL731	1E41-F095	SQ-CL129
		1IA012A	SQ-CL039
0RA026	SQ-CL059	1IA012B	SQ-CL039
0RA027	SQ-CL059	1IA013A	SQ-CL039
0RA029	SQ-CL059	1IA013B	SQ-CL039
0RA029	SQ-CL059		
1B21-F001	SQ-CL227		
1B21-F002	SQ-CL227		
1B33-F019	SQ-CL057		
1B33-F020	SQ-CL057		
1C11-F083	SQ-CL039		
1CM011	SQ-CL419		
1CM014	SQ-CL419		
1CM015	SQ-CL419		
1CM016	SQ-CL419		
1CM017	SQ-CL419		
1CM018	SQ-CL419		
1CM019	**		
1CM022	SQ-CL419		
1CM023	SQ-CL419		
1CM024	SQ-CL419		
1CM025	SQ-CL419		
1CM028	SQ-CL419		
1CM029	**		
1CM031	SQ-CL419		
1CM032	SQ-CL419		
1CM033	SQ-CL419		
1CM034	SQ-CL419		
1CM047	SQ-CL419		
1CM048	SQ-CL419		
1E12-F074A	SQ-CL039		
1E12-F074B	SQ-CL039		
1E12-F301A	SQ-CL047		
1E12-F301B	SQ-CL047		

*Available for review at the Clinton Records Center.

**Manual valve, SQ package not required.

TABLE 3.9-16
BOP Valve/Valve Operator - Seismic Qualification Package
(Q&R MEB (DSER) 76)

VALVE NUMBER	SEISMIC QUALIFICATION PACKAGE*
0VG02YA	SQ-CL265
0VG04VA	SQ-CL265
0VG04YB	SQ-CL265
1E12-F047A	SQ-CL116
1E12-F047B	SQ-CL116
1E12-F052B	SQ-CL107
1E12-F053B	SQ-CL103
1E12-F064A	SQ-CL100
1E12-F087B	SQ-CL109
1E51-F031	SQ-CL068
1E51-F320	**
1IA005	SQ-CL051
1RE021	SQ-CL055
1SA029	SQ-CL052
1SX006C	SQ-CL175
1SX019A	SQ-CL319
1SX019B	SQ-CL319
1SX025A	SQ-CL319
1SX025B	SQ-CL319
1SX025C	SQ-CL319
1VG01YA	SQ-CL265
1VG02YB	SQ-CL265
1VG02YA	SQ-CL265
1VG02YB	SQ-CL265
1VG05YA	SQ-CL265
1VG05YB	SQ-CL265
1VG06YA	SQ-CL265
1VG06YB	SQ-CL265
1VQ03Y	SQ-CL265
1VR006A	SQ-CL156
1VR006B	SQ-CL156
1VR007A	SQ-CL156
1VR007B	SQ-CL156

*Available for review at the Clinton Records Center.

**Manual valve, SQ package not required.

TABLE 3.9-17
(Q&R MEB (DSER) Item 79)
BOP SNUBBER TEST LOADS

PSCo MODEL NO.	PSCo TEST REPORT	RATED LOAD (lb)	TIME AT EACH STEP (sec)	FAULTED LOAD (lb)
PSA 1/4	TR 810	350	11	590
PSA 1/2	TR 811	650	11	1,200
PSA 1	TR 807	1,500	30	2,300
PSA 3	TR 808	6,000	30	11,700
PSA 10	TR 809	15,000	30	23,600
PSA 35	TR 812	50,000	30	91,000
PSA 100	TR 814	120,000	30	190,000

TABLE 3.9-18
(Q&R MEB (DSER) 79)
RATIOS OF MEASURED TO CALCULATED PIPING STRESSES

STRAIN GAUGE DATA SET	STRESS RATIO (MEASURED/CALCULATED)
1	0.53
2	0.55
3	0.73
4	0.48
5	0.59
6	0.55
7	0.57
8	0.70

Therefore, assuming linearity with piping stresses, it is expected that:

$$\frac{\text{Predicted snubber load}}{\text{Calculated snubber load}} = 0.67$$

Since the calculated snubber load is less than the snubber capacity, it can be expected that the predicted snubber load is less than 67% of the snubber capacity.

TABLE 3.9-19
Deleted

ATTACHMENT A3.9

MECHANICAL AND ELECTRICAL COMPONENT DESIGN LOADS

Deleted Table of Contents, List of Tables and List of Figures (pages A3.9-i through A3.9-iv).

MECHANICAL AND ELECTRICAL COMPONENT DESIGN LOADS

A3.9.1 Introduction

The methodologies used to determine the design-basis loads for mechanical and electrical components for the Clinton Power Station (CPS) are presented herein. The methodology for each loading phenomenon is discussed in detail, and the critical load combinations and acceptance criteria are identified. Finally, the system analysis method is discussed.

Section A3.9.2 discusses the methodology for determining safety/relief valve (SRV) actuation loads as they apply to suppression pool boundaries and submerged structures.

Section A3.9.3 discusses the loads resulting from a loss-of-coolant accident (LOCA). Each LOCA-related phenomenon is identified and a loading methodology presented.

Section A3.9.4 identifies the remaining loads that act on mechanical and electrical components i.e., normal, seismic and thermal. Section A3.9.5 identifies the load combinations and acceptance criteria and Section A3.9.6 presents the methods of analyzing the response of the various systems and components of the various loads.

A3.9.2 Development Of SRV Loads

Attachment A3.8 presents the structural design-basis loads for SRV discharge into the suppression pool. The methods used in Attachment A3.8 are based on the assumption that a rams head discharge device would be installed on the SRV discharge lines at CPS. This was consistent with the technical understanding of the phenomena and with the licensing commitments for CPS at the time the construction permit was issued (see Paragraph p. 6-9 of Reference 1). Since this time, General Electric (GE) has determined that the quencher discharge device is a desirable alternative to the rams head device in that it substantially reduces the suppression pool boundary loads resulting from air clearing phenomena during SRV discharge and minimizes thermal effects in the suppression pool. Further, GE has specified the quencher device for the standard BWR/6-238 design and recommends the device for BWR/6-Mark III application (see Attachment A of Reference 2). The quencher device has been incorporated into the CPS design, therefore loads resulting from SRV discharge through this device form the design basis for balance-of-plant (BOP) mechanical and electrical components.

A3.9.2.1 Description of the Phenomena

Prior to the actuation of a pressure relief valve, the downstream piping between the SRV discharge and the suppression pool water surface is full of air at drywell pressure and temperature conditions. The discharge piping terminates at the quencher in the suppression pool, with the water level inside the pipe at the same level as the water level at the pool surface.

When a relief valve lifts, the effluent reactor steam causes a rapid pressure buildup in the discharge pipe. This results in a rapid compression of the column of air in the discharge pipe acceleration of the water in the submerged portion of the pipe and expulsion of the water through the line. The pressure in the pipe builds to a peak as the last of the water is expelled. The compressed cushion of air between the water slug and the steam exits through the quencher and forms a number of clouds of small bubbles which begin to expand to the lower pool pressure. The bubbles continue to expand, displacing the water and propagating a pressure disturbance throughout the suppression pool. When the gas pressure reaches equilibrium with the local hydrostatic pressure, the transient would cease were it not for the inertia of the accelerated water mass. The inertia of the water drives the gas system past the point of equilibrium, and a negative pressure (with respect to local hydrostatic pressure) results within the bubble. The negative pressure in the bubble decelerates the water mass and reverses its motion in an attempt to reach equilibrium. Again the inertia of the water drives the system past the point of equilibrium, and the process repeats in a cyclic manner. The dynamics of the air-water system are manifested in pressure oscillations (similar to that of a spring-mass system) arising from the bubble expansion coupled with inertial effects of the moving water mass. The oscillations are repeated with an identifiable frequency until the bubbles reach the pool surface.

The magnitude of the pressure disturbance in the suppression pool decreases with increasing distance from the point of discharge, resulting in a damped oscillatory load at every point on structures and pool boundaries below the water surface.

A3.9.2.2 Quencher Loads on the Pool Boundaries

A3.9.2.2.1 Pressure on the Drywell Wall, Basemat and Containment Wall

The absolute pressure anywhere on the drywell wall, basemat, and containment wall below the water line may be calculated by the equation:

$$P_o = P_c + \frac{pgh_O}{144g_c} + \sum_{i=1}^{n} \Delta P_{Ri}$$

where:

P_o is the absolute pressure at an arbitrary point on the suppression pool boundary (psia),

P_c is the absolute pressure of the containment atmosphere (psia),

ρ is the density of the suppression pool water (~62.4 lb_m/ft^3),

h_o is the depth of the arbitrary point below the suppression-pool water surface (ft),

g is the local acceleration due to gravity (32.174 ft/sec)

g_c is the standard gravitational constant (32.174 lb_m-ft/lb_f-sec^2),

ΔP_{Ri} is the instantaneous pressure in the i^{th} bubble attenuated to the arbitrary point on the suppression pool boundary, and

n is the total number of bubbles.

The attenuated bubble pressure is a function of the bubble pressure as calculated in Subsection A3.9.2.2.2, and the distance from the discharging quencher to the arbitrary point of application on the suppression pool boundary. Hence,

$$\Delta P_R = 2\Delta P_B\ (t)\left(\frac{r}{r_O}q\right) \qquad \text{for} \quad r_O \ > \ r_q$$

or

$$\Delta P_R = \Delta P_B \quad (t) \quad \text{for} \quad r_O \ \leq \ r_q$$

where:

ΔP_R is the attenuated pressure (psia) at an arbitrary point on the pool boundary due to a bubble with pressure P_B (t),

ΔP_B (t) is the pressure (psig) in a bubble as described in Subsection A3.9.2.2.2 and A3.9.2.2.3,

t is the time (sec),

r_q is the radius of a quencher (4.875 ft), and

r_o is the distance (ft) from the center of the bubble of pressure ΔP_B (t) to the arbitrary point on the pool boundary.

From these equations as presented in Reference 9, it is evident that the following considerations are taken into account in the attenuation model:

a. Bubbles discharged from a quencher device produce a pressure field on the suppression pool boundary that is proportional to the instantaneous bubble pressure an inversely proportional to the distance between the boundary and the bubble.

b. The instantaneous bubble pressure varies with time as described in in Subsection A3.9.2.2.3.

Further assumptions contained within this methodology are as follows:

a. single bubble containing the entire mass of air in the SRV discharge line is formed at the quencher,

b. the bubble is considered to be situated on the centerline of the quencher device, and

c. the bubble has an effective spherical radius equal to the quencher radius (4.875 feet).

A3.9.2.2.2 Peak Bubble Pressures

Peak bubble pressures are calculated by the method described in Subsection A12.6.1 of Reference 2. Table A3.9-1 presents the CPS plant-unique data that were used to determine the peak bubble pressures for the SRV actuation cases considered in Subsection A3.9.2.2.4. Table A3.9-2 presents the maximum positive and negative bubble pressures resulting from the calculation and also presents a comparison of the maximum pressures determined for the

BWR6/238 Standard Plant (Reference 2). The values presented in Table A3.9-1 are conservative for the following reasons:

a. the largest SRV discharge line air volume, VA, has been considered in all discharge cases,

b. the minimum pool surface area, AW, has been used for all discharge cases involving more than two valves,

c. the maximum SRV flow rate, $\dot{M}$, is considered for all discharge cases,

d. the minimum valve opening time, VOT, has been considered for all discharge cases,

e. suppression pool temperatures higher than the normal operating limits are considered for all discharge cases,

f. the containment atmosphere overpressure due to a LOCA for the ADS discharge case is conservatively estimated, and

g. the maximum suppression pool water level is used in the analysis.

As agreed upon in Reference 3, the higher CPS-unique design pressures are used for all SRV discharge lines having an air volume greater than 56.13 ft^3, and the pressure values presented in Reference 2 and Table A3.9-2 are used for SRV discharge lines with air volumes less than 56.13 ft^3.

The pressure margins for the GESSAR-238 design pressures are given in paragraph A5.6 of Reference 2. The pressure margins for the CPS-unique data are presented in Table A3.9-3. This table compares the design pressures with the pressures predicted from the methodology discussed in Reference 2 and quantifies the pressure margin.

Response spectra for the design of BOP piping and equipment have been generated for CPS using the bubble pressures identified and justified in Reference 9 and presented in Table A3.9-2. NSSS piping and equipment have been evaluated against spectra that bound these spectra generated with the bubble pressures presented in Table A3.9-2.

A3.9.2.2.3 Normalized Pressure Time History

An idealized bubble pressure time history is normalized to the maximum positive value, $\Delta P(t)$, as shown in Figure A3.9-1. The frequency of the oscillatory source pressure is 5 to 12 hertz, and the duration of the basic oscillatory loading function is 0.75 second. This frequency and duration accurately reflect the characteristics of the test data given in Attachment A of Reference 2. The frequency of this idealized loading function is adjusted for variation in CPS design parameters discussed in Subsection A3.9.2.2.5.2.3.

It should be noted that the bubble pressure decays to $P_{max}/3$ in five cycles for any frequency in the range of 5 to 12 hertz. For this linear attenuation rule, it is observed that the pressure is fully decayed (i.e., $\Delta P(t) = 0$) in 7.5 cycles after the peak. The justification for this application may be seen in the data presented in Reference 2. In these full-scale plant data, the pressure oscillations are observed to decay to a small fraction of their peak value within 2 or 3 cycles.

Therefore, the consideration of 7.5 cycles of the loading function with the attenuation defined above is conservative.

A3.9.2.2.4 SRV Actuation Cases

SRV discharge piping routed to the suppression pool is arranged so that the points of discharge within the pool are approximately uniformly distributed (see Figure A3.9-2). The location of each valve's discharge around the pool is for distribution of air clearing loads as well as for considerations of pool thermal mixing.

The number of SRV's that can open at one time is dependent on many variables. The following table shows several discrete cases where various numbers of valve openings can be postulated for CPS:

Case	Number of Valves	
(1)	1	Single active failure, normal operator action (first or subsequent actuation)
(2)	2 (adjacent)	1 normal plus single active failure of adjacent valve (first actuation)
(3)	9	All ≥1113-psi setpoint valves (first actuation)
(4)	7	ADS Activation (first actuation)
(5)	16	Vessel pressure ≥1123 psi (first actuation)

The number of SRV's that will open during a reactor vessel pressure transient can be from 1 to 16. This can be shown for situations where various reactor power levels are assumed when the transient event is initiated. Since the discharge points for valves with various setpoints, or those associated with ADS, are distributed around the suppression pool, the discharge of one or two valves represents an asymmetric load on the containment.

A3.9.2.2.4.1 Symmetric and Asymmetric Load Case

The following selected cases represent the asymmetric cases for containment loads:

a. One SRV - This situation can occur due to an operator action or a single active failure. Subsequent actuation of an SRV after an initial pressure transient would be limited to the single 1103-psi setpoint valve.

b. Two adjacent SRV's - This situation can occur due to a pressure transient at low power, which would lift one valve. Concurrent with this the single active failure of an adjacent valve is assumed.

The following selected cases represent the symmetric cases for containment loads:

a. Seven ADS valves - This situation can occur with an intermediate break where the ADS system is activated.

b. Nine valves - This event can occur due to a low power isolation transient.

c. Sixteen (all) valves - This event can occur due to a high power isolation transient.

A3.9.2.2.5 Sequencing of Multiple Valve Discharge Time Histories

This section describes the procedure presented in Reference 9 for determining the SRV discharge 95-95 percent confidence level forcing functions that are imposed on the containment structure to obtain structural responses which are used as input for the evaluation of mechanical and electrical equipment located within the containment. The procedure is different from the structural design-basis procedure because it uses the random nature of several parameters that significantly influence the variable time-phasing relationship of the individual air bubbles formed in the suppression pool during multiple SRV discharge events. The random variables that are used in this procedure are:

a. SRV setpoint tolerance,

b. valve opening time,

c. reactor vessel pressure rise rate, and

d. quencher bubble frequency.

The maximum positive and negative bubble pressures for each individual discharge location are determined by using the method described in Subsection A12.6.1 of Reference 2. It should be noted that the test data on which the peak pressures in Subsection A3.9.2.2.2 are based indicated randomness in the peak pressure amplitude which could also be used for determining structural response. This randomness is ignored here and only 95-95 percent confidence level pressure values are considered to produce a conservative bounding load.

From each of the discharge cases, the Fourier spectra of the forcing functions for 59 Monte Carlo simulations of the event are plotted. A bounding forcing function is then selected in each of the frequency ranges of interest for use in dynamic analysis of the structure.

A3.9.2.2.5.1 Random Parameters

A detailed discussion and justification for the methodology presented below is contained in Reference 9.

A3.9.2.2.5.1.1 Reactor Vessel Pressure Rise Rate (PRR)

The pressure rise rate distribution for BWR/6 plants is shown in Figure A3.9-3. The distribution is determined from an evaluation of BWR/6 transient events. The figure represents the probability density function for pressure rise rates for events opening greater than 11 of the 16 SRV's weighted by the relative occurrence of the events and averaged over all reactor conditions anticipated during the last 40% of an operating cycle. The lower limit of 40 psi/sec is the minimum pressure rise rate expected to open 11 of the 16 SRV's. The upper limit of 140 psi/sec has a high probability of not being exceeded for any operating condition (Reference 9).

It should be noted that the PRR variable is used only in the all-valve case and two lowest-setpoint (nine valve) Monte Carlo event simulations.

A3.9.2.2.5.1.2 Valve Setpoint

The relief setpoints for SRV's on a BWR/6 are arranged in three groups with redundant logic trains consisting of a pressure transducer and three pressure switches. The logic of the 238 BWR/6 design consists of one valve controlled by a pressure switch set at 1103 psi, eight by a pressure switch set at 1113 psi, and the remaining seven by a pressure switch at 1123 psi. A testability feature which utilizes pressure trip instrumentation is also included. The tolerance on the pressure switch setpoints with this testability feature is based on a normal (Gaussian) distribution with a standard deviation of 2 psi as shown in Figure A3.9-4. For the ganged arrangement, the standard deviation is applied to the group setpoints so that the valves within the group will have the same adjustment (Reference 9).

The SRV quencher arrangement and pressure setpoints for CPS are identified in Figure A3.9-2.

A3.9.2.2.5.1.3 Valve Opening Time (VT)

Test data indicate that there is a normal distribution for the VT with a standard deviation of 0.009 second as discussed in Reference 9 and shown in Figure A3.9-5.

A3.9.2.2.5.1.4 Quencher Bubble Frequency Distribution (QBF)

A typical forcing function for a quencher SRV bubble with a frequency of 8 hertz is discussed in Reference 9 and shown in Figure A3.9-1. The bubble lasts approximately 0.75 second in the suppression pool. In the 8-hertz bubble, the pressure decays to one-third of the peak value over five cycles. A complete pressure cycle oscillation period lasts 0.125 second, 0.05 second for the positive pulse and 0.075 second for the negative pulse. For other frequencies, the same damping definition applies, i.e., decay to one-third of the initial value over five cycles, or 0.133 decay per cycle.

The quencher bubble pressure time history in Figure A3.9-1 is an idealized bubble model. For the purposes of this procedure, a pressure time history curve is constructed by assigning half sine waves to both the positive and negative portions. The P_{max} and P_{min} ratios and the positive and negative pulse duration periods are maintained. This provides a time history that is more representative of the test observations (Reference 9) and allows for computer simulation.

Quencher test data show that the frequency of the air bubble is a function of the SRV discharge line air volume. The distribution of bubble frequencies for a discharge line air volume of 50 ft^3 is shown in Figure A3.9-6 and is used as the reference for this procedure. This reference value is the SRV line volume from the operating plants from which the quencher bubble frequency data was obtained. The normal distribution for the curve has a mean frequency of 8.1 hertz with a standard deviation of 1.7 hertz. It is truncated at the minimum and maximum bounds of 5 and 12 hertz (Reference 9).

A3.9.2.2.5.2 Monte Carlo Trial Simulations

A3.9.2.2.5.2.1 Approach

The following four SRV cases as discussed in Reference 9 are considered to produce bounding forcing functions for the equipment evaluations, as discussed in Reference 9:

a. single valve subsequent actuation,

b. two adjacent valves,

c. ADS valves, and

d. all valves.

Case number (3) discussed in Subsection A3.9.2.2.4 is not analyzed because preliminary investigations show that this case is bounded by the all-valve case.

In each of these cases, 59 Monte Carlo trials are performed in which appropriate random variable adjustments are selected for the parameters listed in Subsection A3.9.2.2.5.1. For the single-valve subsequent actuation case only the quencher bubble frequency is varied. For the ADS and two adjacent valve cases, the valve setpoint tolerance and pressure rise rate considerations are not incorporated for obtaining the forcing function. This is because the entire group of ADS valves is simultaneously actuated by a single signal, and in the adjacent valve case simultaneous actuation and failure is assumed. For the all-valve case all variables are considered.

The all-valve trials each consist of selecting a random pressure rise rate from Figure A3.9-3 and a random pressure switch setpoint for each group of SRV's using Figure A3.9-4. This information is used to compute the bubble arrival time difference or separation between the groups of valves. These bubble arrival times are adjusted for each individual valve by randomly selecting a valve opening time (VT) using Figure A3.9-5.

Once the bubbles are in the suppression pool, each bubble frequency is randomly varied by selecting a frequency from a unique distribution for the discharge line volume involved (see Figure A3.9-6). The bubble pressure time history for each valve location is then used to determine the forcing function on the suppression pool boundary by utilizing the methods described in Subsection A3.9.2.2.1.

For the ADS and two adjacent valve cases, it is assumed that all valves receive the opening signal at the same instant and then bubble phasing is adjusted by randomly selecting a different VT for each valve. Each bubble frequency is then randomly selected as for the multiple valve trials.

A3.9.2.2.5.2.2 <u>Bubble Arrival Time</u>

A3.9.2.2.5.2.2.1 <u>Calculation of Reference Arrival Time</u>

The arrival time for each air bubble in the suppression pool relative to the lowest setpoint SRV is a function of the SRV setpoint arrangement and the reactor pressure rise rate. Assuming no tolerance on setpoints, no variation in valve opening time (VT), and randomly selecting a pressure rise rate (PRR), the arrival times of the bubbles in the suppression pool are computed by dividing the nominal setpoint differences (i.e., Δ p = 10 and 20 psi for BWR-6) by the PRR. It should be noted that SRV discharge line lengths are not considered as discussed in Reference 9. For BWR-6 with nominal setpoints at 1103, 1113, and 1123 psi, the time separation is 0.077 and 0.154 second, based upon PRR = 130 psi/sec.

A3.9.2.2.5.2.2.2 Adjustment of Bubble Arrival Time for Pressure

Each all-valve Monte Carlo trial includes an adjustment of the bubble arrival times as calculated above by slightly increasing or decreasing the valve setpoint for each group of valves. This is done by using a random number generator code to select valve setpoint variation from the distribution presented in Reference 9 and shown in Figure A3.9-4.

A3.9.2.2.5.2.2.3 Adjustment of Bubble Arrival Time for Valve Opening Time Variations

Each Monte Carlo trial includes an adjustment of the bubble arrival time as calculated above by slightly increasing or decreasing the VT for each valve. This is done by using a random number generator code to select VT variation from the distribution presented in Reference 9 and shown in Figure A3.9-5.

A3.9.2.2.5.2.3 Quencher Bubble Frequency Variation

A3.9.2.2.5.2.3.1 Adjustment of Bubble Frequency for Discharge Line Air Volume

As indicated in Subsection A3.9.2.2.5.1.4 the frequency of the quencher bubble is a function of the SRV discharge line air volume. A reference line air volume of 50 ft^3 has been selected to generate the bubble pressure time history shown in Figure A3.9-1. For each SRV discharge line volume a unique frequency distribution is generated by adjusting all of the characteristics (mean, standard deviation, lower bound, upper bound) of the reference distribution curve by multiplying by the cube root of the ratio of 50 ft^3 to the actual air volume in the SRV discharge line as discussed in Reference 9. For example, the adjustment of frequency for a line volume of 65 ft^3 is:

$$8.1 \ \ \text{Hz} \times \frac{50}{65} = 8.1 \times 0.92 = 7.4 \ \ \text{Hz}$$

Examples for the other characteristics:

Vol.	Mean	Std. Dev.	Lower Bound	Upper Bound
(ft^3)	(Hz)	(Hz)	(Hz)	(Hz)
50	8.1	1.7	5	12.0
65	7.4	1.6	4.6	11.0

A3.9.2.2.5.2.3.2 Adjustment of Quencher Bubble Time History for Selected Frequency

In each Monte Carlo trial, a random number generator code is used to select a frequency from each of the frequency distribution curves generated above. For each frequency selected, a time history of the quencher bubble pressure oscillation is generated by adjusting the reference time history shown in Figure A3.9-1, as discussed in Reference 9. This is accomplished by maintaining the ratio of negative to positive pulse period constant. The pressure cycle period, positive pressure pulse time and negative pressure pulse time are adjusted by multiplying each by the ratio of the reference frequency (8 hertz) to the selected frequency. For example, for 6 hertz:

$$\text{Pressure cycle period} = 0.125 \text{ sec} \frac{8 \text{ Hz}}{6 \text{ Hz}} = 0.167 \text{ sec}$$

Positive pressure pulse time =

$$0.05 \text{ sec} \frac{8 \text{ Hz}}{6 \text{ Hz}} = 0.067 \text{ sec}$$

Negative pressure pulse time =

$$0.075 \text{ sec} \frac{8 \text{ Hz}}{6 \text{ Hz}} = 0.100 \text{ sec}$$

Number of cycles per 0.75 sec duration =

$$\frac{\text{Bubble duration}}{\text{Pressure cycle period}} = \frac{0.75 \text{ sec}}{0.167 \text{ sec / cycle}} = 4.5 \text{ cycles}$$

A3.9.2.2.5.3 Factors Affecting Pressure Distribution on the Suppression Pool Boundary

A3.9.2.2.5.3.1 Bubble Pressure Attenuation

The attenuation of the bubble pressure with distance r from the quencher is $2r_q/r_o$, where r = radius of the quencher (4.87 ft), and $\geq 2r_o$ (see Subsection A3.9.2.2.1). The distance, r_o, is the true spatial distance from the quencher center to the node (Reference 9).

A3.9.2.2.5.3.2 Line-of-Sight Influence

The line-of-sight criterion for the bubble pressure states that points which are not in a direct line from the outer radius of the quencher arms to the location in question will not be affected by the pressure from the quencher (Reference 9).

A3.9.2.2.5.3.3 Combination of Multiple SRV Pressure Time Histories

The time sequencing application provides a given phase relationship between quencher bubbles. The pressure at each node point and time step is calculated by combining the contribution from each valve (in the line of sight) using algebraic summation (Reference 9). At each node where the total calculated pressure at any time step exceeds the maximum pressure (positive or negative) from any of the contributing valves, the calculated pressure at the specific time step is set equal to the maximum bubble pressure at the same instant in time. This adjustment is made to assure that the conservative pressure calculation technique does not violate potential flow theory.

A3.9.2.2.5.4 Bounding Forcing Functions for Plant Evaluation

A3.9.2.2.5.4.1 Time Sequencing

Time sequencing with random parameters is used to arrive at the forcing function for the multiple SRV air-clearing events referenced in Subsection A3.9.2.2.5.2.1.

A Monte Carlo technique is used to generate the building forcing function for equipment evaluations. The bounding forcing function from 59 trials will result in a 95% confidence level that 95% of the time the actual forcing function will be less than the forcing function determined by the Monte Carlo technique (Reference 9).

A3.9.2.2.5.4.2 Pressure Time Histories

The criteria stated above require fifty-nine trials of pressure distribution on the pool boundary which are calculated using the random parameters delineated in Subsection A3.9.2.2.5.1.

A3.9.2.2.5.4.3 Vertical Basemat Force and Overturning Moment

The total basemat force is calculated as a function of time by integrating the node pressures over the suppression pool basemat incremental areas. The overturning moments (about two perpendicular horizontal axes through the basemat center upper surface) are calculated, as a function of time, by integrating the product of node pressure x the incremental area moment arm x the incremental area over the suppression pool boundary (containment, basemat, and drywell wall).

A3.9.2.2.5.4.4 Fourier Spectra

Fourier spectra (see References 4 and 5) of the vertical basement force and overturning moment discussed above for the 59 trials are developed for the selected cases used to determine dynamic responses for equipment evaluations. The significant frequency range is divided into three frequency intervals as recommended in Reference 9 and determined below:

Step 1. Adjust the mean frequency of each SRV discharge line for air volume differences (see Subsection A3.9.2.2.5.2.3.1).

Step 2. Calculate the mean frequency (f_m) for all applicable SRV discharge lines.

Step 3. Establish the frequency intervals based on 0.5 f_m to 1.5 f_m, 1.5 f_m to 2.5 f_m, and 2.5 f_m to 3.5 f_m.

$$\text{where } f_m = \frac{1}{N} \sum f_i ;$$

$i = 1, \ldots N$, and

N = total number of valves actuated.

The basemat loading trials with the largest spectral value within each frequency interval (from the 59 trials) are selected for determination of bounding forcing functions for plant assessment.

A3.9.2.2.5.5 Structural Response Analysis

Forcing functions corresponding to the trials selected in each frequency range are used as input to the structural analysis described in Subsection A3.8.6. The resulting dynamic responses are then enveloped for plant evaluation.

A3.9.2.3 Quencher Loads on Submerged Structures

This section provides various examples of safety relief valve (SRV) quencher discharge loads on piping systems in the suppression pool.

A3.9.2.3.1 Submerged Structures Selected to Illustrate SRV Loads

Quencher SRV discharge loads are presented for the following piping system components and supports:

a. Main steam safety relief valve (MSRV) line and quencher arm

b. Non-MSRV line

  1. Deleted

  2. RCIC return line and support

The MSRV lines and quenchers are located near the drywell wall while the RCIC return line is near the containment wall. These are shown schematically in Figures A3.9-9, A3.9-10, A3.9-11, and A3.9-12. Also shown in the figures are the nodes for which loads are presented in this section. These nodes are not necessarily those with the highest SRV loads, but were selected to provide examples of the SRV loads on variously oriented structures near the inner and outer boundaries of the suppression pool.

A3.9.2.3.2 Loading Phenomena and Calculation Approach

The loading phenomena have their beginnings in the transient in the discharge line following actuation of a safety relief valve, as described in Subsection A3.9.2.1. This transient results in water, compressed air and then steam being expelled from the discharge line into the suppression pool through a quencher. Quencher water clearing loads on submerged structures are not applicable to Clinton-I submerged structures per Subsection 3BL.3.1 of Reference 10. The compressed air discharged through the quencher, however, produces significant loads on structures in the pool. It is assumed this compressed air produces four spherical bubbles, each located between a different pair of quencher arms (per Subsection 3BL.3.2 of Reference 10). As described in Subsection A3.9.2.1, bubbles oscillate in the pool as they are overexpansed and recompressed during their rise to the pool surface. This produces pressure oscillations throughout the pool, resulting in dynamic loads on submerged structures. The pressure potential field used in calculating these dynamic loads is determined by the method of images, as discussed in Subsection A3.8.2.2.2, and accounts for the rigid boundaries and free surface of the annular three dimensional pool in determining pressure potential attenuation factors. These attenuation factors and the time dependent bubble strengths define the pressure distribution within the pool. The pressure gradient determines the pool acceleration, which with the virtual mass of a structure, determine its inertial load in a potential flowfield. Integration of the acceleration determines velocity, from which standard drag and lift loads are determined. For a particular structure, appropriate inertial drag, standard drag and lift coefficients were evaluated (References 11 through 17), and used, in conjunction with pool acceleration and velocity to determine time histories of loads on that structure. SRV loads on the toroidal ECCS suction strainer are determined utilizing the methodology specified in GESSAR II. CPS specific bubble pressures and maximum SRV line volume have been used to determine the oscillating air bubble characteristics.

A3.9.2.3.3 Examples of Quencher Air Discharge Loads on Submerged Structures

A3.9.2.3.3.1 Nodalization of Structures

Nodes for example quencher air discharge loads are shown on Figures A3.9-10, A3.9-11 and A3.9-12. Node diameters and lengths are also shown in these figures. Node lengths are not the same for structures of different diameter. In general, node lengths were optimized to minimize computer time while still maintaining the desired accuracy. All other things being equal, larger nodes have larger loads. Inertial drag loads are proportional to nodal volume, while standard drag and lift loads are proportional to the projected area of the node normal to the flow. When comparing loads in this section for the various submerged structures, these comparisons are then made in light of the different nodal dimension.

A3.9.2.3.3.2 Selection of Safety Relief Valve Actuation Cases

SRV actuation cases considered are given in Subsection A3.9.2.2.4. Results presented here include only those SRV discharge cases that produced an extreme load component in at least one direction. For some structures with identical geometry , but different azimuthal locations, more than one discharge case of the same type was postulated to ensure that extreme load components were determined.

A3.9.2.3.3.3 Quencher Air Discharge Load Results

Examples of quencher air discharge loads on submerged piping components listed in Subsection A3.9.2.3.1 are presented in Table A3.9-8. Unsteady flow and interference effects due to submerged structures and boundary proximity were considered (References 11 through 17). For each node, results are presented for the SRV discharge cases that produced extreme load components on that node, or on another node on the same structure. Load components are identified as either horizontal or normal for Nodes 1 and 3. For Nodes 6 through 9 only resultant loads are presented. Node numbers referred to in Table A3.9-8 as well as load component direction are defined in Figures A3.9-10 through A3.9-12. Resultant load directions are defined in Table A3.9-8. Example time histories for Nodes 1 and 3 for various quencher air discharge cases are presented in Figures A3.9-13 through A3.9-14.

A3.9.3 Development Of LOCA Loads On BOP Equipment

Chapter 6 and Attachment A3.8 provide a detailed description of the LOCA phenomena in a BWR/6 Mark III containment. NSSS and BOP equipment are affected by these loads both directly and indirectly. The direct-effect loads result from the reaction of the suppression pool and drywell response to LOCA phenomena on the equipment. The indirect effects result from the containment response to the action of LOCA phenomena on the containment boundaries. The direct effects are discussed in Subsections A3.9.3.1 through A3.9.3.5. The indirect effects are discussed in Attachment A3.8 and Section 3.8. These effects are incorporated into equipment design, as described in Section A3.9.4.

A3.9.3.1 Vent Clearing Water Jet Loads

A3.9.3.1.1 Indirect Effects

The indirect effects of vent clearing water jet loads are evaluated in Subsection A3.8.6.2.

A3.9.3.1.2 Direct Effects

Water jets from the LOCA vents during the vent clearing transient induce a flowfield throughout the suppression pool. This flowfield indirectly creates dynamic loads in addition to those due to the direct jets, on structures in the suppression pool, (Subsection 3BL.2.2 of Reference 10), but the indirect loads are bounded by the LOCA air bubble loads. In the Clinton Power Station, there are no submerged structures (other than the main steam SRV quencher devices) in the direct path of the vent clearing water jets. Therefore, the LOCA air bubble loads are used conservatively in place of the water jet load.

A3.9.3.2 Vent Clearing Air Bubble Loads

A3.9.3.2.1 Indirect Effects

The indirect effects of vent clearing air bubble loads are evaluated in Subsection A3.8.6.2.

A3.9.3.2.2 Direct Effects

After the water clearing transient, pressurized drywell air goes through the vents, and a single bubble is formed around each top vent. As these bubbles grow, unsteady fluid motion is created within the suppression pool, and all submerged structures below the pool surface will be exposed to transient hydrodynamic loads. Using the procedure outlined in Section 3BL.2.3 of Reference 10, LOCA air bubble loads were computed for the structures listed in Subsection A3.9.2.3.1. These loads include a conservatism factor of 2 to cover the effects of a moving source. Unsteady flow and interference effects were considered per References 6, 7, 11, 12, 13, 14, 15, and 16. This results in increased loads and conservativeness not explicitly mentioned in GESSAR-II. Results of these calculations at various node points on those submerged structures are presented in Table A3.9-9. Time history at one of the selected node points is shown in Figure A3.9-23.

A3.9.3.3 Pool Swell Impact and Drag Loads

The expanding air bubbles cause the suppression pool surface to rise until the bubble breaks through the surface. Structures within the suppression pool located at or above the elevation of the bottom vent will be subject to drag loads during the pool swell transient. Furthermore, structures above the pool surface, within the region of pool swell, will be subject to both drag and impact loads due to: (1) the assumed bulk motion of the pool, (2) the froth formed during and after the LOCA air bubble breaks through the water, and (3) the fallback transient for the pool water (Reference 9).

These loads for the structures listed in Subsection A3.9.2.3 were calculated using the methodology described in Subsections A3.9.3.3.1 and A3.9.3.3.3. Drag and fallback loads are presented in Tables A3.9-10 and A3.9-11, respectively.

A3.9.3.3.1 Drag Load Methodology

The pool swell and froth swell drag loads are based on the use of classical standard and acceleration drag equations. The bounding velocity is based on the following equation:

$$\frac{V}{V_{max}} = \frac{H}{10}\left(2.6 - 1.6\sqrt{\frac{H}{10}}\right) \qquad \text{for } H < 10$$

where V_{max} = 50 feet per second

V = velocity at a given elevation (ft/sec)

H = height above the initial pool surface (ft)

For $H \geq 10$, $V = V_{max}$. The density of water is used in determining the pressure differential across structures in the bulk pool swell region.

The density of 18.8 1bm/ft^3 used in the froth drag calculations (Reference 9) was determined by assuming that the water contained in the top 9 feet of the suppression pool and the containment air between the HCU floor and the suppression pool are homogeneously mixed. This density is deemed to be conservative, since PSTF 1/3-scale test data indicate that the density is approximately 10 1bm/ft^3.

Additional considerations in the computation of pool swell drag loads are the interference effects due to local flow perturbations that might significantly affect the flow field and lift load, which might arise due to vortex shedding. These interference effects are accounted for in the methodology discussed in References 6 and 7.

A3.9.3.3.2 Impact Load Methodology

Structures above the suppression pool are subject to impact loads resulting from the pool swell transient. The normalized profile of this load is presented in Figure A3.9-7 and is based on the data presented in Reference 2.

The load is applied to all structures within the bulk pool swell regions, as defined in Reference 8 and depicted in Figure A3.9-8. The structures within the froth swell region are designed to the load profile shown in Figures A3.8-18.

A3.9.3.3.3 Fallback Load Methodology

After bulk pool swell has stopped and breakthrough has occurred, the water will fall back into the suppression pool. The fallback loads are estimated by the methodology discussed in Subsection A3.9.3.3.1. The velocity used in these calculations is 35 ft/sec, the terminal velocity for a 20-foot free fall (from the HCU floor elevation).

A3.9.3.4 Condensation Oscillation Loads

A3.9.3.4.1 Indirect Effects

The indirect effects of condensation oscillation loads are evaluated in Subsection A3.8.6.2.

A3.9.3.4.2 Direct Effects

Steam condensation begins after the vent is cleared of water and the drywell air has been carried over into the wetwell and induces bulk water motion which creates drag loads on structures submerged in the pool. Using the procedure outlined in Subsection 3BL.2.6 of Reference 10, condensation oscillation loads were calculated for the structures listed in Subsection A.3.9.2.3. Results of these calculations at various node points on those submerged structures are presented in Table A3.9-12. As noted in Table A3.9-12, the forcing function is approximately a sine wave with frequency of 2 to 3.5 Hz and zero to peak amplitude as shown.

A3.9.3.5 Chugging Loads

A3.9.3.5.1 Indirect Effects

The indirect effects of chugging loads are evaluated in Subsection A3.8.6.2.

A3.9.3.5.2 Direct Effects

Chugging occurs as the drywell air is being purged and the vent mass flux falls below a critical value. Chugging then induces acoustic pressure loads on structures submerged in the pool. The procedure for calculating these loads and obtaining their time histories is described in Subsection 3BL.2.8 of Reference 10.

A time window of 0.002 seconds was used in determining the number of sources that contributed to the chugging load and 4000 ft/sec acoustic velocity in water was assumed. Loads calculated according to this procedure, at various node points on the submerged structures listed in Subsection A3.9.2.3.1, are presented in Table A3.9-13. Chugging loads are to be modeled as having a magnitude equal to the resultant loads in Table A3.9-13 and a duration of .002 seconds. The wave shape is assumed to be square and the period between individual chugs is 1 to 5 seconds. Chugging loads on the toroidal ECCS suction strainer are calculated utilizing the acoustic chugging methodology.

A3.9.4 Development Of Other Loads For BOP Piping And Equipment

The following additional loads on piping and equipment are considered in the design basis of those components subject to suppression pool hydrodynamics. These loads are applied to the piping and equipment in combination with those suppression pool hydrodynamic loads of the applicable load combination(s) (see Tables A3.9-6 and A3.9-7).

A3.9.4.1 Loads for BOP Piping

Pressure

Stress is induced in the pipe by the internal pressure on the pipe wall.

Weight

The sustained load consists of the weight of pipe, pipe fittings, pipe contents, and insulation where applicable.

Thermal

A secondary, self-limiting load results from the piping system anchors and restraints to thermal expansion.

Design pressure, temperature, and weight information is contained in the system design specification for each piping subsystem.

Hydraulic Transient

A dynamic load on applicable systems due to appreciable and sudden changes in the mass flow rate in the piping system, caused by sudden valve opening or.closure, pump trip or pump startup.

Valve opening transient analysis is performed for the main steam SRV discharge lines and the RHR heat exchanger relief valve discharge lines to the suppression pool. Development of hydraulic transient loading input for the main steam SRV lines is described in Subsection 3.9.3.3.1 and is generally applicable to the RHR pressure relief valve discharge lines.

Hydraulic transient loads are also considered for two other cases: feedwater pump trip and main steam stop valve closure.

Seismic

The development of seismic input for subsystem analysis is described in Subsection 3.7.1. A description of subsystem modelling and analysis techniques is contained in Subsection 3.7.3.

The use of combined response spectra input in the design basis is discussed in Subsection A3.9.5.

A3.9.5 LOAD COMBINATIONS

Tables A3.9-6 and A3.9-7 form the basis for load combinations and acceptance criteria for BOP systems and components. Each subsystem is analyzed for the governing load combination(s) for each service level. The response spectra curves for the dynamic loads within each service level shown in Table A3.9-6 are combined to generate a set of upper bound response spectra.

The upper bound spectra loads and applicable hydrodynamic loads, (i.e., loads derived in a manner other than by response spectra) for the specified load combinations are combined at the response level. The method of combination for dynamic loads is the SRSS method except as noted in Table A3.9-6.

Differential support/anchor displacement effects of the various dynamic loads are also considered in the analysis of pipe stresses and component support loads. Functional capability requirements as outlined in the attachment of the table have been considered.

A3.9.6 ANALYSIS METHODS

A3.9.6.1 Analysis Methods - BOP Piping

Analysis of BOP piping is performed with the computer program PIPSYS, a linear, elastic, three-dimensional space frame, finite-element program. The piping system is idealized using two node beam elements, each with six degrees of freedom. The system is defined by its spatial configuration and geometric and material properties. The system is modelled with straight elements, curved elements, or springs. Spring hangers, snubbers and rigid restraints can be specified.

The program performs static (including thermal) and dynamic analysis along with response spectrum, forced vibration time history, and earthquake time history analyses and computes the combined stresses based on Section III of the ASME Boiler and Pressure Vessel Code.

For additional information on the program, see FSAR Subsection 3.9.1.2.6.9.

VESLFAT is a program that has been used in ASME Section III, Subsection NB-3200 evaluation of BOP piping.

A3.9.6.1.1 Weight and Thermal

The program uses a stiffness approach to analyze systems subjected to uniform and/or concentrated weight, thermal loads, and prescribed boundary displacements.

A3.9.6.1.2 Hydraulic Transient

The hydraulic transient analysis of relief valve discharge piping is performed in two steps. The computer program SRVA is used to generate transient forcing functions, as described in USAR Subsection 3.9.3.3. The analysis method of SRVA is briefly outlined in USAR Subsection 3.9.1.2.6.17.

The forcing functions time history is then input to PIPSYS in a forced vibration time history analysis (see USAR Subsection 3.9.3.3.1). The mode acceleration method or the direct integration method is used. The characteristics of fluid transient forcing functions are such that higher modes, including several axial modes, have significant contribution, necessitating the consideration of a larger number of modes. The mode acceleration method computes dynamic response in terms of the static solution plus a connection to the static solution.

The hydraulic transient forcing function generation for feedwater pump trip and main steam stop valve closure is performed using the computer program HYTRAN (Subsection 3.9.3.3). The forcing functions are utilized in the PIPSYS analysis similar to relief valve discharge transient analysis.

A3.9.6.1.3 Seismic

As described in Subsection A3.9.5, a set of upper limit design response spectra has been generated for the response spectra input of applicable dynamic loads (including seismic OBE and SSE) and governing load combinations.

In the response spectrum method, the input acceleration is described in terms of a response spectrum, which is a plot of the maximum response of a single degree of freedom oscillator with different natural frequencies to a given input acceleration and structural damping.

The analytical procedures of the response spectrum method of piping analysis are discussed in USAR Subsection 3.7.3.8.

A3.9.6.2 Analysis Methods and Criteria for BOP Equipment

For the analysis and design of equipment and their supports, all the loading combinations shown in Tables A3.9-6 and A3.9-7 shall be considered. The seismic loads shall be conservatively combined with the pool dynamic loads using the SRSS method of combining loads.

The acceptance criteria used for BOP equipment are shown in Table A3.9-5 for both active and nonactive equipment and for fluid and nonfluid system components.

Nonactive fluid system equipment shall be reviewed for the same loading combinations and the corresponding ASME Section III design limits. Operability of all active components shall be established by performing a detailed deformation analysis or alternatively by performing a prototype test.

Operability of Class IE electrical equipment shall be verified by performing a prototype test or alternatively by a method using a combination of test and analysis.

A3.9.7 References

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2. "Information Report - Mark III Containment Dynamic Loading Conditions," General Electric Company Report NED0-11314-08, submitted as Amendment 37 to 238-NI-GESSAR, December 1975 as revised through Amendment 43 in Appendix 3B.

3. Illinois Power Company Letter No. U-0018, L. J. Koch to K. Kniel, dated September 23, 1976. Subject: Acceptance Criteria for Quencher Loads - Clinton Power Station.

4. Cooley, J. W., and Turkey, J. W., (1965), "An Algorithm for the Machine Calculation of Complex Fourier Series," Mathematics of Computation, Vol. 19, No. 90, pp. 297-301.

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9. General Electric Co., Appendix 3B of 238-NI-GESSAR, submitted GESSAR docket STN 50-447, dated March 31, 1980, application for final design approval.

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14. Zdravkovich, M. M., "Review-Review of Flow Interference Between Two Circular Cylinders in Various Arrangements," J. of Fluids Engineering, 618-633 (December 1977).

15. Patton, K. T., "Tables of Hydrodynamic Mass Factors for Translational Motion," ASME Publication 65-WA/UNT-2.

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17. Sarpkaya, T., Garrison, C. J., "Vortex Formation and Resistance in Unsteady Flow," J. of Applied Mechanics, 16-24 (March 1963).

18. Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria, Generic Technical Activity A-8, NUREG-0487, Supplement No. 2, U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington D.C., February 1981.

19. Mark II Improved Chugging Methodology, NEDE-24822-P Class III General Electric Company, May 1980; This document was prepared for the Mark II Utility Owners' Group by Bechtel Power Corporation under contract with General Electric Company, (Proprietary).

20. Mark II Containment Program Load Evaluation and Acceptance Criteria, Generic Technical Activity A-8, NUREG-0808, U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington D.C., August, 1981.

21. An Approach to Chugging, Assessment of RHR Steam Discharge Condensation Oscillation in Mark III Containments, Prepared for the Mark III Containment Issues Owners' Group, Job 16031, Bechtel Power Corporation, San Francisco Power Division, Nuclear Engineering Staff, March 1984, (Proprietary).

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TABLE A3.9-1
INPUT DATA FOR PEAK BUBBLE PRESSURE CALCULATIONS
FOR CLINTON POWER STATION*

PARAMETER	UNITS	CASE 1	CASE 2	CASE 3	CASE 4 ***	CASE 5	CASE 6
VA	ft^3	64.411	64.411	64.411	64.411	64.411	64.411
AQ	ft^2	74.622	74.622	74.622	74.622	74.622	74.622
AW	ft^2	7174.6	7174.6	3587.3	395.4	395.4	395.4
$\dot{M}$	lb_m/sec	318.5	318.5	318.5	318.5	318.5	318.5
TW	°F	100.0	120.0	100.0	100.0	100.0	120.0
WCL	ft	17.82	17.82	17.82	17.82	17.82	17.82
VØT	msec	20.0	20.0	20.0	20.0	20.0	20.0
ZQ	ft	13.92	13.92	13.92	13.92	13.92	13.92
DP	psig	0.0	0.0	0.0	0.0	0.0	5.0
VAAQ	ft	0.255	0.255	0.255	0.255	0.255	0.255
MNAQ MNQJ	**	11.483	11.483	11.483	11.483	11.483	11.483
MNQ1	**	6.89	6.89	6.89	6.89	6.89	6.89
MNQ2	**	47.472	47.472	47.472	47.472	47.472	47.472
LNTW	**	3.632	3.890	3.632	3.632	3.632	3.890
WCL2	ft^2	29.502	29.502	29.502	29.502	29.502	29.502
AWAQ	-	20.0	20.0	20.0	5.296	5.296	5.296
AWQ2	-	400.0	400.0	400.0	28.046	28.046	28.046
PINF	bar	1.430	1.430	1.430	1.430	1.430	1.774
PRD1	bar	0.669	0.704	0.669	0.880	0.880	0.915

*The parameters presented in this table are the input, intermediate and final results for the methodology of Section A12 of Reference 2.
**The dimensions of these computed variables do not have physical meaning.
***This is not a design-controlling load case.

TABLE A3.9-2
DESIGN PRESSURES COMPARED WITH
GESSAR-238 DESIGN PRESSURES

	GESSAR-238[1]		CPS[2, 3]	
	P_B^+	P_B^-	P_B^+	P_B^-
CASE DESCRIPTION	(psid)*		(psid)*	
1. Single valve – first actuation	10.8	-6.5	11.9	-7.0
2. Single valve – subsequent actuation	18.3	-7.8	20.2	-8.1
3. Adjacent valves – first actuation	10.8	-6.5	11.9	-7.0
4. Low setpoint and next low setpoint valves - first actuation[4]	10.9	-6.0	12.5	-6.5
5. All valves – first actuation	12,1	-6.4	12.5	-6.5
6. ADS valves – first actuation	11.3	-6.8	13.1	-7.3

*Does not include atmospheric pressure or hydrostatic head

1. Based on maximum SRV line air volume of 56.13 ft#
2. Based on a maximum SRV line air volume of 65.0 ft#.
3. Based on the variables specified in Table A3.9-1.
4. Not a design-controlling load case.

TABLE A3.9-3
ESTIMATED MARGINS IN PEAK BUBBLE PRESSURES FOR CPS

	DISCHARGE CASE*					
	1	2	3	4	5	6
Design pressures based on Clinton-unique parameters (psid) (+/-)	11.9/-7.0	20.2/-8.1	11.9/-7.0	12.5/-6.5	12.5/-6.5	13.1/-7.3
Predicted maximum bubble pressure, (psid) (+/-)	7.8/-5.3	6.3/-4.8	7.8/-5.3	8.3/-5.1	8.3/-5.1	8.6/-5.7
Pressure margin, (psi) (+/-)	4.1/1.7	13.9/3.3	4.1/1.7	4.2/1.4	4.2/1.4	4.5/1.6
% margin (based on design pressures) (+/-)	35/25	69/41	35/25	34/21	34/21	34/22

*Discharge cases correspond to those listed in Table A3.9-2

Table A3.9-4

Deleted

TABLE A3.9-5
LOAD COMBINATIONS AND ALLOWABLE STRESS LIMITS FOR BOP EQUIPMENT

A. Nonfluid System Equipment

PLANT CONDITION	ACTIVE (EQUIPMENT) ELASTIC DEFLECTION	NONACTIVE AND ACTIVE (EQUIPMENT) IN-ELASTIC DEFLECTION
Upset		
(Normal Operating Loads + Load Cases 1 through 3 of Table A3.9-6)	$\sigma_m \leqq 0.6\ S_y$ (D.M.) $\leqq 0.3\ S_u$ (B.M.)	$\sigma_m \leqq 0.6\ S_y$ (D.M.) $\leqq 0.4\ S_u$ (B.M.)
	$\sigma_t \leqq 0.7\ S_y$ (D.M.) $\leqq 0.4\ S_u$ (B.M.)	$\sigma_t \leqq 0.9\ S_y$ (D.M.) $\leqq 0.6\ S_u$ (B.M.)
Faulted		
(Normal Operating Loads + Load Cases 4 through 15 of Table A3.9-6	$\sigma_m \leqq 0.7\ S_y$ (D.M.) $\leqq 0.4\ S_u$ (B.M.)	$\sigma_m \leqq 0.9\ S_y$ (D.M.) $\leqq 0.6\ S_u$ (B.M.)
	$\sigma_t \leqq 0.95\ S_y$ (D.M.) $\leqq 0.6\ S_u$ (B.M.)	$\sigma_t \leqq 1.5\ S_y$ (D.M.) $\leqq 0.9\ S_u$ (B.M.)

TABLE A3.9-5 (Cont'd)

B. Active Fluid System Equipment

PLANT CONDITION	ASME CLASS 1	ASME CLASS 2 & 3
Upset		
(Load Cases 1 through 3 of Table A3.9-6 + Normal Operating Loads)	Per ASME Sec. III Same as Nonactive	Per ASME Sec. III Same as Nonactive
Emergency/Faulted		
(Load Cases 4 through 15 of Table A3.9-6 + Normal Operating Loads)	$\sigma_m \leqq 1.00\ S_m$ $\sigma_t \leqq 1.5\ S_m$	$\sigma_m \leqq 1.00\ S$ $\sigma_t \leqq 1.65\ S$

where:

σ_m = Membrane stress

σ_t = Membrane + bending stress

S_m, S = As defined by Section III

S_y = Yield stress at corresponding temperature

S_u = Ultimate stress at corresponding temperature

D.M. = Ductile material

B.M. = Brittle material

TABLE A3.9-6
LOAD COMBINATIONS TABLE
FOR SAFETY-RELATED PIPING AND COMPONENT SUPPORTS

				SRV			SBA	IBA	DBA (Note 2)							Stress Level (Note 1)	Method of Combination for Dynamic Loads
Load Case	N	SSE	OBE	1V2 P	ADS	ALL	CH	CO/CH	AP	WJ	MVC	PS	FB	CO	CH		SRSS (Note 3)
1	X															A	N/A
2	X		X			X										B	SRSS
3	X		X	X												B	SRSS
4	X	X				X										C	SRSS
5	X	X		X			X									C	SRSS
6	X	X		X				X								C	SRSS
7	X	X			X		X									C	SRSS
8	X	X			X			X								C	SRSS
9	X	X							X	X						C	SRSS
10	X	X								X						C	SRSS
11	X	X									X					C	SRSS
12	X	X									X	X				C	SRSS
13	X	X											X			C	SRSS
14	X	X											X	X		C	SRSS
15	X	X													X	C	SRSS

GENERAL TABLE NOTES

1. Use of Stress Level D is allowed for some systems and load combinations as outlined in the system design specification. All ASME Class 1, 2, and 3 piping systems which are required to function for safe shutdown under the postulated events shall meet criteria of NEDO-21985 "Functional Capability Criteria for Essential Mark II Piping," September, 1978. Stress Levels shown are only applicable for piping and component supports designed by loading rating. Equipment stress levels are shown in Table A3.9-5.

2. Main vent clearing to be added to pool swell for the case of submerged structures only.

3. Use of SRSS for combination of dynamic loads complies with and is demonstrated by the guidelines of NUREG-0484, Rev. 1.

TABLE A3.9-7*

LOAD COMBINATIONS GOVERNED BY CORRESPONDING STRESS LEVEL COMBINATIONS OF TABLE A3.9-6

				SRV			SBA	IBA	DBA (Note 2)							Stress Level (Note 1)	Method of Combination for Dynamic Loads
Load Case	N	SSE	OBE	1V2P	ADS	ALL	CH	CO/CH	AP	WJ	MVC	PS	FB	CO	CH		SRSS (Note 3)
16	X		X													B	N/A
17	X			X												B	N/A
18	X					X										B	N/A
19	X			X			X									C	SRSS
20	X				X		X									C	SRSS
21	X				X			X								C	SRSS
22	X					X	X									C	SRSS
23	X	X														C	N/A
24	X	X		X												C	SRSS
25	X			X				X								C	SRSS
26	X					X		X								C	SRSS
27	X		X		X		X									C	SRSS
28	X		X		X			X								C	SRSS
29	X								X	X						C	SRSS
30	X									X						C	N/A
31	X										X					C	N/A
32	X										X	X				C	SRSS
33	X												X			C	N/A
34	X												X	X		C	SRSS
35	X														X	C	N/A

*The General Table Notes for Table A3.9-6 are also applicable to this Table.

TABLE A3.9-8
EXTREME QUENCHER AIR DISCHARGE LOADS ON SUBMERGED STRUCTURES
AT VARIOUS NODE POINTS

NODE* NO.	DISCHARGE CASE	SUBMERGED STRUCTURE	^{F}HORIZONTAL* ($1b_f$)	^{F}NORMAL* ($1b_f$)	^{F}RESULTANT ($1b_f$)	TIME HISTORY FIG. NO.
1	Single valve subsequent actuation	MSRV line	0	-179.4	-	A3.9-13
	All valve	MSRV line	0	-164.8	-	-
	Asymmetric (two adjacent)	MSRV line	-41.2	-123.8	-	-
3	Single valve subsequent actuation	Quencher arm	-136.3	342.1	-	A3.9-14
	All valve	Quencher arm	-153.2	-265.4	-	-
	Asymmetric (two adjacent)	Quencher arm	-183.4	-227.3	-	-
4	Deleted					

TABLE A3.9-8 (Cont'd)

NODE* NO.	DISCHARGE CASE	SUBMERGED STRUCTURE	^{F}HORIZONTAL* ($1b_f$)	^{F}NORMAL* ($1b_f$)	^{F}RESULTANT ($1b_f$)	TIME HISTORY FIG. NO.
5	Deleted					
6	Single valve subsequent actuation	Hydrogen sparger	N/A	N/A	N/A	-
	All valve	Hydrogen sparger	-	-	393.9**	-
	ADS	Hydrogen sparger	-	-	251.9**	-
	Asymmetric (two valve)	Hydrogen sparger	-	-	260.2**	
7	Single valve subsequent actuation	Hydrogen sparger	N/A	N/A	N/A	-
	All valve	Hydrogen sparger	-	-	64.1**	-
	ADS	Hydrogen sparger	-	-	70.2**	-
	Asymmetric (two adjacent)	Hydrogen sparger	-	-	64.1**	-

TABLE A3.9-8 (Cont'd)

NODE* NO.	DISCHARGE CASE	SUBMERGED STRUCTURE	^{F}HORIZONTAL* ($1b_f$)	^{F}NORMAL* ($1b_f$)	^{F}RESULTANT ($1b_f$)	TIME HISTORY FIG. NO.
8	Single valve subsequent actuation	RCIC return line & support	-	-	29.9***	-
	All valve	RCIC return line & support	-	-	29.0***	-
	ADS	RCIC return line & support	-	-	33.7***	-
	Asymmetric (two adjacent)	RCIC return line & support	-	-	36.3***	-
9	Single valve subsequent actuation	RCIC return line & support	-	-	43.8**	-
	All valve	RCIC return line & support	-	-	132.0**	-
	Asymmetric (two adjacent)	RCIC return line & support	-	-	54.4**	-
	ADS	RCIC return line & support	-	-	79.9**	-

NOTES:
1. Extreme component loads do not necessarily occur at the same time.
2. N/A = not applicable.
3. - = not available.

*See Figures A3.9-10, A3.9-11, and A3.9-12 for load component directions and node numbers.
**Resultant lies in vertical plane normal to axis of structure.
***Resultant lies in horizontal plane normal to axis of structure.

TABLE A3.9-9
EXTREME LOCA AIR BUBBLE LOADS ON SUBMERGED STRUCTURES
AT VARIOUS NODE POINTS

NODE* NO.	SUBMERGED STRUCTURE	^{F}HORIZONTAL* ($1b_f$)	^{F}NORMAL* ($1b_f$)	^{F}RESULTANT ($1b_f$)	TIME HISTORY FIG. NO.
1	MSRV line	44	181	-	A3.9-23
2	Quencher arm	371	329	-	-
4	Deleted				
5	Deleted				
6	Hydrogen sparger	-	-	539.6**	-
7	Hydrogen sparger	-	-	156.1**	-
8	RCIC return line & support	-	-	24.4***	-
9	RCIC return line & support	-	-	238.4**	-

NOTES:
1. Extreme component loads do not necessarily occur at the same time.
2. - = not available.

*See Figures A3.9-10, A3.9-11, and A3.9-12 for component directions and node numbers.
**Resultant lies in vertical plane normal to structure.
***Resultant lies in horizontal plane normal to structure.

TABLE A3.9-10
POOLSWELL DRAG LOADS ON SUBMERGED STRUCTURES
AT VARIOUS NODE POINTS

NODE* NO.	SUBMERGED STRUCTURE	^{F}VERTICAL* (1b$_{f}$)	DURATION (sec)	COMMENTS
1	MSRV line	N/A	-	See Note 1
2	Quencher arm	N/A	-	See Note 1
4	Deleted			
5	Deleted			
6,7	Hydrogen sparger	-8210	-	Load on entire sparger configuration
8	RCIC return line	N/A	-	Vertical pipe
9	RCIC return line support	-10406	-	Load on entire support strut

NOTES:
1. LOCA air bubble phase is terminated when air bubble engulfs the structure. At that time water has already moved past structure. Therefore, no poolswell loads.
2. N/A = not applicable.
3. - = not available.

*See Figures A3.9-10, A3.9-11, and A3.9-12 for load directions and node numbers.

TABLE A3.9-11
FALLBACK DRAG LOADS ON SUBMERGED STRUCTURES
AT VARIOUS NODE POINTS

NODE *NO.	SUBMERGED STRUCTURE	^{F}VERTICAL* ($1b_f$)	DURATION (sec)	COMMENTS
1	MSRV line	331	0.229	Applied normal to axis of structure
2	Quencher arm	1955	0.076	
4	Deleted			
5	Deleted			
6	Hydrogen sparger	4040	-	Load on entire sparger configuration
7	Hydrogen sparger	4040	-	
8	RCIC return line	N/A	-	Vertical pipe
9	RCIC return line support	5130	-	Load on entire support strut

NOTES:

1. N/A = not applicable.
2. - = not available.

*See Figures A3.9-10, A3.9-11, and A3.9-12 for load directions and node numbers.

TABLE A3.9-12
EXTREME CONDENSATION OSCILLATION LOADS ON SUBMERGED STRUCTURES
AT VARIOUS NODE POINTS

NODE** NO.	SUBMERGED STRUCTURE	^{F}HORIZONTAL** ($1b_f$)	^{F}NORMAL** ($1b_f$)	^{F}RESULTANT ($1b_f$)	TIME HISTORY* FIG. NO .
1	MSRV line	0	7	7	-
2	Quencher arm	24	12	27	-
4	Deleted				
5	Deleted				
6	Hydrogen sparger	-	-	25.3***	-
7	Hydrogen sparger	-	-	8.7***	-
8	RCIC return line & support	-	-	2†	-
9	RCIC return line & support	-	-	8.7***	-

NOTE:
1. - = not available.
*Sine wave with frequency of 2 to 3.5 hz and zero to peak amplitude as shown in table above, or forcing function equation in Reference 10, Page 3B-20, Subsection 3B.4.1-5 and Attachment F.
**See Figures A3.9-10, A3.9-11, and A3.9-12 for component directions and node numbers.
***Resultant lies in vertical plane normal to structure.
†Resultant lies in horizontal plane normal to structure.

TABLE A3.9-13

EXTREME CHUGGING LOADS ON SUBMERGED STRUCTURES AT VARIOUS NODE POINTS

NODE** NO.	SUBMERGED STRUCTURE	^{F}HORIZONTAL** ($1b_f$)	^{F}NORMAL** ($1b_f$)	^{F}RESULTANT ($1b_f$)	TIME HISTORY* FIG. NO.
1	MSRV line	-2	108	108	-
2	Quencher arm	125	248	278	-
4	Deleted				
5	Deleted				
6	Hydrogen sparger	-	-	14.3***	-
7	Hydrogen sparger	-	-	9.1***	-
8	RCIC return line & support	-	-	86.9†	-
9	RCIC return line & support	-	-	4.8***	-

NOTE:
1. - = not available.
*Square wave, period between chugs equal 1 to 5 seconds, Reference 10, Page 3B1-24.
**See Figures A3.9-10, A3.9-11, and A3.9-12 for component load directions and node numbers.
***Resultant lies in vertical plane normal to structure.
†Resultant lies in horizontal plane normal to structure.

Attachment B3.9

GENERAL ELECTRIC CO.
SAN JOSE, CALIFORNIA
PLANT PIPING DESIGN
DESIGN MEMO #123-8402
DESIGN RECORD FILE DRF NO. B21-00042

VERIFICATION CONTAINED IN DRF

REPORT ON NRC QUESTION ABOUT THERMAL GRADIENT STRESSES FOR CLINTON X-QUENCHER

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Plant Piping Design

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Plant Piping Design

Q&R 210.02

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B3.9.1 Introduction.

Clinton X-Quencher had been analyzed according to the requirements of the Subsection ND-3600 of the ASME Code, Section III. The NRC had expressed some concern about the fatigue life of the X-Quencher due to thermal gradient and local bending stresses. Although ND-3600 does not require a detailed fatigue evaluation (and the intent of ND-3600 had been met in the previous analysis), it was decided to reanalyze four critical locations (shown on Figure B3.9-1) on the X-Quencher to the requirements of Subsection NB-3600 of ASME Code to alleviate NRC's concern on the subject. The analysis consisted of calculating the fatigue usage factor at these locations for 40 years of the plant life.

B3.9.2 Loads And Method.

The four locations (designated as 'A', 'B', 'C' and 'D') represent the critical sections. Location 'A' is the SRV piping - quencher interface point and was selected because it is a dissimilar metal connection. Location 'B' is the X-Quencher adapter - reducer interface connection. Location 'C' is the quencher arm - body interface connection and location 'D' is the quencher connection to the pedestal adaptor. Fatigue usage factor was calculated for these locations using peak stress 'S_p' and alternating stress 'S_a' obtained from Equations 11 and 14 of Subsection NB-3600 respectively. To calculate the worst thermal gradients across these sections, the most severe transient was used. The transient consists of a step change from 70°F to 350°F. Appropriate heat transfer coefficients were calculated and applied for the transient. The stresses due to these thermal gradients were combined with the stresses due to the moment loading at these locations and the pressure stress. Secondary stress indices C_1, C_2 and C_3 and local stress indices K_1, K_2 and K_3 were calculated and applied for these locations based on their configurations.

A finite element model was made for location 'D' because of its complex geometric configuration and the consequent difficulty in calculating stress indices. Peak stress was calculated using this model for the same thermal transient, the moment loading and the pressure loading.

B3.9.3 Results And Conclusions.

The fatigue usage factors calcuated are shown below.

LOCATION	USAGE FACTOR
‘A’ SRV PIPING SIDE	0.84
‘A’ QUENCHER SIDE	0.32
B	0.21
C	0.53
D	0.01

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Attachment B3.9 (Cont'd)

The thermal gradients calculated are shown below:

LOCATION	ΔT_1*	ΔT_2*	ΔT_A*	T_B*	TIME** MINUTES
A	120.838	126.180	94.956	109.951	0.010
B	52.430	183.841	79.405	79.405	0.020
C	118.026	97.114	95.083	154.196	0.240
D	By Finite Element Analysis.				

* Refer to Subsection NB-3600 for definition.
** The time which creates maximum combined stresses.

Since the usage factors for the four critical sections are less than 1.0, we can safely conclude that the Clinton X-Quencher can withstand all normal and upset condition loads including the stresses due to thermal gradient and local bending effects.

B3.9.4 References

(1) ASME Boiler and Pressure Vessel Code Section III - 1983 Edition.

(2) DRF #B21-42, X-Quencher, Pedestal Mounted.

(3) Stress Report Data Sheet for X-Quencher #22A5408AB.

(4) Design Specification for Quencher X-Type #21A2139.

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ATTACHMENT C3.9
Program Descriptions for TRIPE, EASE2, E2A17 and EWELD
(Q&R 210.04)

C3.9.1 TPIPE

COMPUTER PROGRAM: TPIPE Version 5.1 Dated 6/23/83

AUTHORS: PMB Systems Engineering
San Francisco, CA

DESCRIPTION:

A special purpose computer program using finite element scheme to perform static and dynamic linear elastic analyses of power related piping system. Dynamic analysis options include:

1. Frequency Extraction
2. Response Spectrum
3. Time History Modal Superposition
4. Time History Direct Integration

ADDITIONAL CAPABILITIES INCLUDE:

1. Plot undeformed and deformed geometry.

2. Post process pipe member end forces through ASME Section III Class 1, 2, or 3 Stress Evaluation Equations, and provide support load and pipe attachment evaluation.

3. Thermal transient heat analysis to provide linear thermal gradient, T1 nonlinear thermal gradient, T2, and gross discontinuity expansion difference.

The major computational algorithms which solve the linear equilibrium equations and calculate the dynamic structural frequencies and mode shapes were taken from the efficient General Purpose Structural Analysis Program SAPIV.

EXTENT OF APPLICATION:

TPIPE has been utilized to perform the following category of design analysis for the CRD Hydraulic System piping.

1. Static
2. Frequency Analysis
3. Response Spectrum
4. Time History Direct Integration
5. ASME Class II and B31.1 Code Evaluation

VERIFICATION

The verification for the program has been done by PMB Systems Engineering. A total of thirty-six (36) test cases were utilized. Portions of the program are compared with hand calculations. The dynamic analysis was compared with results from accepted computer programs.

In addition the seven (7) NRC Benchmarks referred within Item II.2 or C.(5) in SRP 3.9.1 NUREG-0800 have been executed through TPIPE.

REFERENCED COMPUTER PROGRAMS:

PISOL	--	EDS Nuclear
NUPIPE	--	Control Data Corp.
STARDYNE	--	Control Data Corp.
SAP IV	--	Control Data Corp.

C3.9.2 EASE2

COMPUTER PROGRAM: EASE2 Version 13.4 Dated 4/10/82

AUTHORS: Engineering Analysis Corporation
Berkeley, CA

SOURCE: Control Data Cybernet Services
Sunnyvale, CA

PROGRAM DESCRIPTION:

EASE2 is a general purpose computer program that uses finite element scheme to analyze linear elastic static and dynamic models. The element library consists of beams, pipe, isoparametric solids, isoparametric 2-D elements, shell, plate and membrane. EASE2 is capable of performing the following analyses:

1. Static
2. Eigenvalue
3. Mode Superposition
4. Direct Integration
5. Response Spectrum

EASE2 employs a modified Gauss elimination procedure in banded equation block solver for static analysis. The static analysis model (K-Matrix) is fully compatible with the dynamic analysis method.

EXTENT OF APPLICATION:

EASE2 has been utilized at RCI For the design analysis of finite element models of support structures for the CRD piping. The options used are:

1. Static
2. Eigenvalue Extraction
3. Direct Integration
4. Response Spectrum

VERIFICATION:

The EASE2 has a substantial user base. This program has been used by various nuclear engineering facilities. EASE2 has been verified by its authors. The test cases are published as the EASE2 Example Problems Manual by Fred Peterson Engineering Analysis Corporation.

Specifically the program was verified in the following three ways:

1. Theoretical results published in text books.
2. Numerical results produced by other previously verified computer program.
3. Hand calculations.

EXTENDED COMPUTER PROGRAMS:

1. SAP IV University of California, Berkeley
2. ANSYS Swanson Analysis Systems Inc., Elizabeth, PA.

C3.9.3 E2A17

COMPUTER PROGRAM: E2A17 Version 13.4B Dated 8/22/83

AUTHORS: Engineering Analysis Corporation
Berkeley, CA

DESCRIPTION:

E2A17 is a post-processing program that uses the geometry and end forces and moments from the EASE2 program and performs steel design check calculations according to the provisions of Section III Nuclear Power Plant Requirements ASME Boiler and Pressure Vessel Code, Article XVII-2000, Linear Elastic Analysis, 1977 (1979 Winter Addenda).

EXTENT OF APPLICATION:

E2A17 is utilized in evaluating results provided by "EASE2" in the analysis of support structures for the CRD piping.

VERIFICATION:

E2A17 has been verified against hand calculations.

C3.9.4 EWELD

COMPUTER PROGRAM: EWELD Version 2.0 Dated 10/03/83

AUTHORS: Ramji Chaudhari, Simon Schmukler and Kent Johnson

DESCRIPTION:

EWELD is a post-processing program that uses the forces and moments at element end-nodes and the section properties data to calculate fillet weld thickness, lamellar tearing and flare bevel weld sizes.

EXTENT OF APPLICATION:

EWELD is utilized as a post-processor to the E2A17 computer programs, to analyze weld size and tabulate reactions as a step in the design analysis of CRD piping support structures.

VERIFICATION:

EWELD program has been verified against hand calculations.

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Seismic qualification includes the equipment qualification due to applicable hydrodynamic and LOCA loads which are addressed in Attachment A.3.9. This section addresses the dynamic qualification of all Class 1E electrical equipment, instrumentation and their supports. Section 3.9 addresses the dynamic qualification of safety-related mechanical equipment and Section 3.11 addresses the environmental qualification of Class 1E mechanical and electrical equipment.

All Class 1E mechanical and electrical equipment and instrumentation are designed to withstand, without functional impairment, the effects of the Safe Shutdown Earthquake (SSE) defined in Subsection of 3.7.1 and the hydrodynamic loads discussed in Attachment A.3.9.

The requirements of IEEE-344 and Regulatory Guide 1.100 are met for equipment identified in this section.

Per Section 6.1.1 of IEEE Standard 344-75, electrical equipment must be tested on a shake table with mounting and configuration similar to actual service, unless adequate justification can be made to extend the qualification to an untested orientation or configuration. Safety-related switchgears in Div. 1, 2 and 3 Auxiliary Power Systems were seismically qualified with breakers in racked-in/racked-up/connected configurations to meet the requirements of IEEE-344-75. However, to address certain breaker configurations other than the original qualification (which may exist during on-line maintenance activities of the switchgear breakers), evaluations are performed to provide the required justification to extend the qualification to certain untested configurations.

The switchgears addressed in this evaluation are safety-related Div. 1 and 2, 4160V/6900V switchgear, Div. 3 HPCS switchgear and 480V Unit Substation switchgear in the Unit Auxiliary power systems.

The identification of all Class IE BOP and NSSS electrical equipment and instrumentation utilized in various systems of the plant is provided in Nuclear Station Engineering Standard MS-02.00. The information provided in MS-02.00, Maintenance of Equipment Qualification Program Manual is as follows:

a. Equipment Number: Provides the specific number and name of the equipment. This provides a correlation between other documents and drawings.

b. Equipment Manufacture/Model: Identifies the manufacturer or vendor of the equipment and equipment model number.

c. Qualification Package: List the specific document package which demonstrates qualification.

Equipment functional times, environmental zones, and equipment categories can be obtained from the respective environmental qualification document package referenced in MS-02.00.

3.10.1 Seismic Qualification Criteria

3.10.1.1 BOP and NSSS Compliance with IEEE-344

BOP and NSSS Class IE electrical equipment and instrumentation is qualified to meet the requirements of IEEE-344. The dynamic qualification and its documentation is verified to show that the equipment performs its function during and after the SSE event. This event is combined with any applicable hydrodynamic event as discussed in Section A3.9. Analysis, testing, or a combination of test and analysis is used to qualify the equipment. The method of qualification is identified in the qualification document package referenced in MS-02.00.

An analysis or test or a combination of test and analysis is used for qualifying Class IE electrical equipment. Analytical methods when used are consistent with Section 5.0 of IEEE-344. Testing is primarily used for qualifying Class IE electrical active equipment. The equipment tested is in compliance with Section 6.0 of IEEE-344.

When the equipment could not be practically qualified by methods using analysis or testing alone because of its size and/or complexity, a method of combined analysis and testing was used.

3.10.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

3.10.2.1 BOP Equipment

All Class IE electrical equipment and instrumentation are identified in Nuclear Station Engineering Standard MS-02.00. These components are designed to withstand the effects of the SSE, hydrodynamic and LOCA loads as applicable, without functional impairment.

For each unique piece of instrumentation or electrical equipment a dynamic qualification report has been prepared by the equipment vendor in accordance with the requirements of the equipment procurement specifications. The load combinations used for BOP equipment qualification are those delineated in Tables A3.9-6 and A3.9-7. The equipment is qualified in the entire frequency range of interest to include the higher frequencies (greater than 33 Hz) for the hydrodynamic and LOCA events. For those components qualified by testing, the requirements of IEEE-344 are verified or adequate justification is provided that the component does meet these requirements.

Since several components/devices are used at different locations for the same application or for a different application, the approach taken for qualification is to test the component for the bounding or worst case location and for the different applications. For example, a relay in one system may have as its safety function to deenergize and open its contact within a certain time, while in another system it may be required to energize and close its contacts. In such a situation, the relay would be tested in both modes under the worst case dynamic conditions to assure operability.

To the extent practical, dynamic qualification tests of equipment were performed while the equipment was subjected to normal operating loads. Where it was demonstrated by prior testing or analysis that operating loads such as pressure, torque, flow, voltage, current, and thermal expansion did not cause significant stress loads within the equipment, or where such operating loads are not significant to equipment operability, operation under such loads during testing was not required. The equipment was monitored and evaluated during and after the test for malfunction or failure and, upon completion of the test, was thoroughly inspected for damage. The results of such tests were documented and reviewed for compliance with IEEE-344.

Testing was used for the seismic qualification of equipment auxiliary components, such as relays, switches, and instruments necessary for proper operation. As far as possible, these components were tested and qualified with the equipment mounted in a manner similar to the field mounting condition. The input motion was applied simultaneously to the vertical axis and one principal horizontal axis, or when single axis tests were performed, adequate justification was provided. The maximum input motion acceleration was equal to or in excess of the maximum dynamic acceleration at the equipment mounting. The qualification document package referenced in Nuclear Station Engineering Standard MS-02.00 gives specific details on methods, results, and analysis.

In some cases where it was found that different pieces of equipment have similar characteristics, the test program was based upon testing of prototype equipment. The test reports furnished by the equipment supplier were reviewed for compliance with IEEE-344.

3.10.2.2 NSSS Equipment

All Class IE NSSS instrumentation and electrical equipment are identified in Nuclear Station Engineering Standard MS-02.00. These components are designed to withstand the effects of the SSE, hydrodynamic and LOCA loads as applicable, without functional impairment.

For each unique piece of instrumentation or electrical equipment, a dynamic qualification report has been prepared and assessed to verify compliance with IEEE-344 and the loading combination as defined in Table 3.9-2. The qualification shall also be verified in the entire frequency range of interest to include the higher frequencies (greater than 33 Hz) of the hydrodynamic and LOCA events, or adequate justification is provided that the component does meet the requirements.

The dynamic loading criterion used in the design and subsequent qualification of all Class IE instrumentation and electrical equipment supplied by GE is as follows:

The Class IE equipment shall be capable of performing all safety-related functions during (1) normal plant operation, (2) anticipated transients, (3) design basis accidents, and (4) postaccident operation while being subjected to, and after the cessation of, the accelerations resulting from the SSE or hydrodynamic loads at the point of attachment of the equipment to the building or supporting structure.

The criteria for each of the devices used in the Class IE systems depend on the use in a given system; for example, a relay in one system may have as its safety function to deenergize and open its contacts within a certain time, while in another system it must energize and close its contacts. Since GE supplies many devices for many applications, the approach taken was to test the device in the worst case configuration in which it might be used. In this way, the capability of protective action initiation and the proper operation of safety-failure circuits is assured.

From the basic input ground motion data, a series of response curves at various building elevations are developed after the building layout is completed. Standard requirement levels that meet or exceed the maximum expected unique plant information is included in the purchase specifications for Seismic Category I equipment. Suppliers of equipment such as batteries and racks, instrument racks, control consoles, etc., are required to submit test data, operating experience and/or calculations to substantiate that their components, systems, etc. will not suffer loss of function during or after dynamic loadings. The magnitude and frequency of the loadings which each component will experience are determined by its specific location within the plant.

All Class IE electrical equipment and instrumentation are evaluated for their capability to perform the safety function under the combinations of seismic and hydrodynamic vibration loadings shown in Table 3.9-2.

3.10.2.3 NSSS Testing Procedures for Qualifying IE Electrical Equipment and Instrumentation (Excluding Motors and Valve-Mounted Equipment)

The test procedure required that the devices be mounted on the table of the vibration machine in a manner similar to that in which it is installed. The device was tested in the operating states in which it is to be used when performing its Class IE functions and these states were monitored before, during, and after the test to assure proper function and absence of spurious function. In the case of relays, both energized and deenergized states and normally open and normally closed contact configurations were tested if the relay is used in those configurations for its Class IE functions.

The first step was to search for resonances in each device. This was done since resonances cause amplification of the input vibration and is the most likely cause of malfunction. The resonance search was usually run at low acceleration levels (0.2g) in order to avoid damaging the test sample in case a severe resonance was encountered. The resonance search was generally run up to 60 Hz to account for high frequency effects. If the device was large the vibrations were monitored by accelerometers placed at critical locations from which accelerations were measured and compared with the input acceleration level at the table to determine resonance.

The method used for qualification is a dynamic excitation with a single sinusoidal frequency with peak acceleration amplitude at several discrete frequencies. The vibratory excitation was applied in three orthogonal axes individually.

Additionally, after conducting the frequency scan and resonance determination, the devices were tested to determine their malfunction limit. This test was a necessary adjunct to the assembly test as explained below. The malfunction limit test was run at each resonant frequency as determined by the resonance search. In this test, the acceleration level was gradually increased until either the device malfunctioned or the limit of the vibration machine was reached. If no resonances were detected (as was usually the case), the device was considered to be rigid, and the malfunction limit was therefore independent of frequency. To achieve maximum acceleration from the vibration machine, rigid devices were malfunction-tested at the upper test frequency of 60 Hz, since that allowed the maximum acceleration to be obtained from deflection-limited machines.

The above procedures were required of purchased devices as well as those supplied by GE. Vendor test results were reviewed and if unacceptable, the tests were repeated either by GE or the vendor. If the vendor tests were adequate, the device was considered qualified to the limits of the test.

3.10.2.4 Qualification of Valve Mounted Equipment - NSSS

The piping analyses establish the response spectra, power spectral density function or time history characteristics, and horizontal and vertical accelerations for the pipe-mounted equipment. Class IE motor-operated valve actuators were qualified per IEEE-382.

The safety/relief valves, including the electrical components mounted on the valve, are subjected to a dynamic test. This test is described in Subsection 3.9.3.2.1.5.2.

3.10.2.5 Qualification of NSSS Motors

Seismic qualification of the ECCS motors is discussed in Subsection 3.9.2.2.1.6.7 in conjunction with the ECCS pump and motor assembly. Seismic qualification of the Standby Liquid Control pump (SLC) motor is discussed in Subsection 3.9.2.2.1.6.10 in conjunction with the SLC pump motor assembly.

3.10.3 Methods and Procedure of Analysis of Testing of Supports of Electrical Equipment and Instrumentation

3.10.3.1 BOP Seismic Category I Electrical Equipment and Instrument Supports

3.10.3.1.1 Battery Racks, Instrument Racks, Control Consoles, Cabinets and Panels

Response spectra curves at the appropriate locations, consisting of the response due to seismic, hydrodynamic and LOCA loads (where applicable) have been supplied to the equipment vendor.

Methods used for qualification are analysis, testing, or a combination thereof in accordance with the procedures in Subsection 3.10.2.1. The qualification results for BOP components is presented in the qualification document package referenced in Nuclear Station Engineering Standard MS-02.00.

3.10.3.2 NSSS Dynamic Analysis, Test Procedures, and Restraint Measures

3.10.3.2.1 Instrument Racks, Control Consoles, Cabinets, and Panels

Class IE electrical equipment supplied by GE is used in many systems on many different plants under widely varying dynamic loading requirements. The seismic qualification tests were generally performed at all frequencies from 5 to 60 Hz. The actual qualification range was generally 1 to 60 Hz. However, since test facility capability was sometimes limited, the lower frequency tested was 5 Hz. A combination of test and analysis was used to assure that all component resonances were determined.

Some GE-supplied Class IE devices were qualified by analysis only. Analysis was used for passive mechanical devices and was sometimes used in combination with testing for larger assemblies containing Class IE devices. For instance, a test might have been run to determine if there were natural frequencies in the equipment within the critical loading frequency range. If the equipment was determined to be free of natural frequencies within the critical frequency range, then it was assumed to be rigid and a static analysis was performed as shown in Attachment A3.10. If it had natural frequencies in the critical frequency range, then calculations of transmissibility and responses to varying input accelerations were determined to see if Class

IE devices mounted in the assembly would operate without malfunctioning. A sample analysis is shown in Attachment B3.10. In general, the testing of Class IE equipment was accomplished using the procedure described in the following paragraph.

Assemblies (i.e., control panels) containing devices which have had dynamic malfunction limits established were tested by mounting the assembly on a vibration machine in the field-mounted configuration as far as practical. Whenever exceptions to this were identified, additional justification was provided. A low-level resonance search was then conducted. As with the devices, the assemblies were tested in the three major orthogonal axes. The resonance search was run in the same manner as described previously for devices. If resonances were present, the transmissibility between the input and the location of each Class IE device was determined by measuring the accelerations at each device location and calculating the amplification between it and the input. From the transmissibilities the response at any Class IE device location for any given input was determined analytically. (It was conservatively assumed that the transmissibilities were linear as a function of acceleration even though they actually decrease as acceleration is increased.) If the device input accelerations were determined to be below their malfunction limits, then the assembly was considered a rigid body with a transmissibility equal to one so that a device mounted on it would be limited directly by the assembly input acceleration.

There are basically three generic panel types. One or more of each type was tested using the above procedures.

Figures 3.10-1 through 3.10-3 illustrate the three basic panel types and show typical accelerometer locations. The results of the dynamic tests on the Class IE panels supplied by GE are presented in the qualification document package referenced in Nuclear Station Engineering Standard MS-02.00.

The full acceleration level tests described above demonstrated that most of the panel types had more than adequate mechanical strength and that a given panel design acceptability was just a function of its amplification factor and the malfunction levels of the devices mounted in it. Subsequent panels were, therefore, tested at lower acceleration levels and the transmissibilities measured to the various devices as described above. By dividing the malfunction levels of the devices by the panel transmissibility, between the device and the panel input, the panel seismic qualification level could be determined. Several high-level tests have been run on selected generic panel designs to assure conservatism in using the transmissibility analysis described.

In cases where the supports or panels are not separable from the components being qualified, an integrated method of testing is performed with the component mounting fastened to the test table in a manner identical to the actual installation. Thus a qualification procedure in accordance with that outlined in Subsection 3.10.3.2.1 is used.

3.10.3.3 Design of Cable Trays, Cable Tray Supports, and Conduit Supports

3.10.3.3.1 General

All safety-related cable trays, their hangers, conduits, and their supports are designed to meet the requirements of Seismic Category I electrical components.

3.10.3.3.2 Loads

a. Dead load D: includes dead weight of the cables, cable trays, conduits and self weight of the hanger.

b. Live Load L: a live load of 200 pounds is considered for the design of cable trays and cable tray hangers, for the construction loading case only.

c. E: operating-basis earthquake or safe shutdown earthquake, whichever is larger.

d. Safety-relief valve discharge load.

SRV_{ALL}	=	SRV loading due to 16 (all) safety/relief valve discharge.
SRV_{1V2P}	=	SRV loading due to one safety/relief valve subsequent actuation.
SRV_{ADS}	=	SRV loading due to seven (ADS) safety/relief valve discharge.

e. LOCA dynamic response loads P_d.

MVC	=	LOCA loading due to main vent clearing.
PS	=	LOCA loading due to pool swell.
CO	=	LOCA loading due to condensation oscillation.
CH	=	LOCA loading due to chugging.
AP	=	LOCA loading due to annulus pressurization and associated pipe breaks.

3.10.3.3.3 Load Combinations and Design Limits

a. Cable Tray Supports

D	1.0 x allowable
D* + L	1.33 x allowable but not to exceed 0.95 F_y (D* tray weight only without cables)

$$\left.\begin{array}{l} D+E+SRV_{ALL} \\ D+E+SRV_{1V2P}+CH \\ D+E+SRV_{1V2P}+CO \\ D+E+SRV_{ADS}+CH \\ D+E+SRV_{ADS}+CO \\ D+E+MVC \\ D+E+AP \\ D+E+PS \end{array}\right\}$$

1.6 x allowable, but a minimum factor of safety of 1.05 will be maintained against yield

b. Conduit Supports

D	1.0 x allowable
$D + E + SRV_{1V2P}$ $D + E + SRV_{ALL}$ $D + E + SRV_{1V2P} + CH$ $D + E + SRV_{1V2P} + CO$ $D + E + SRV_{ADS} + CO$ $D + E + SRV_{ADS} + CH$ $D + E + SRV_{1V2P} + MVC$ $D + E + SRV_{1V2P} + PS$ $D + E + SRV_{1V2P} + AP$	1.6 x allowable, but less than or equal to $0.95F_y$

3.10.3.3.4 <u>Procedure for Analysis and Design</u>

The dynamic analysis and design of the cable tray hangers is performed using computer programs PIPSYS and SEISHANG. Both the programs utilize a response spectrum method of analysis. Different dynamic loads are combined by the square root of the sum of the squares method, with the exception of the condensation oscillation load, which is combined by absolute sum. The stresses and reactions from the different directional excitations are combined by the square root of the sum of the squares method.

The equivalent static analysis of the cable trays is performed using the computer program SEISHANG and peak of the response spectrum.

Conduit supports are also designed using equivalent static approach and peak of the response spectrum.

The PIPSYS program performs a multimode analysis. Stresses and reactions from all significant modes are combined using methods in compliance with NRC Regulatory Guide 1.92. The SEISHANG program performs a single mode analysis. Seven percent damping was used in the analysis for SSE.

The slenderness ratio for compression members shall be as follows:

Member Type	Maximum Slenderness Ratio (kl/r)
Compression members (verticals, diagonals and longitudinal braces) in floor and wall mounted supports (i.e., compression system supports)	200
Compression members (verticals, diagonals and longitudinal braces) in ceiling mounted supports (i.e., tension system supports)	300

A detailed discussion of these programs is given in Appendix C.

3.10.3.3.5 Applicable Codes, Standards and Specifications

a. AISI "Specification for Design of Cold-formed Steel Structural Members," 1968 Edition and 1980 Edition.

b. AISC "Specification for the Design Fabrication and Erection of Structural Steel for Buildings," (1969 or 1978).

c. AWS D1.3, "Structural Welding Code - Sheet Steel," (1978 Edition).

d. AWS D1.1, "Structural Welding Code - Steel."

Clarifications and exceptions to AWS D1.1 and D1.3 are made based on engineering evaluations.

3.10.3.3.6 Instrument Tubing Supports

The dynamic design and analysis of instrument tubing supports are in accordance with the simplified dynamic analysis discussed in Subsection 3.7.3.8.6.

ATTACHMENT A3.10
SAMPLE SEISMIC STATIC ANALYSIS

ATTACHMENT A3.10
SAMPLE SEISMIC STATIC ANALYSIS

PART I

Part I presents a set of curves from which static seismic analysis of standard enclosures can be quickly performed. A standard enclosure is any enclosure listed in the Enclosure Standards Manual. The enclosures are assumed to be floor mounted, using all mounting holes with 5/8 inch steel bolts or studs each having an effective area of 0.2256 in2. Using an elastic limit of one half the ultimate strength, the bolts are assumed to have a maximum safe tension stress and maximum safe shear stress of 28,000 PSI and 21,000 PSI, respectively. The curves are based on a design basis earthquake having a horizontal acceleration of 1.5G and a vertical acceleration of 0.5G. It is assumed that each enclosure is mounted alone and not coupled directly to any other enclosure.

The static analysis consists of determining the maximum allowable safe weight of the enclosure and its components for which the mounting bolt stresses are not exceeded. The curves of Figure A3.10-1 have been derived for this purpose. To use the curves given in Figure A3.10-1, first determine from Table A3.10-1 the curve designation of the enclosure being considered. Next, using the corresponding curve in Figure A3.10-1, determine the maximum safe weight per bolt for a given height of the center of gravity. The maximum safe enclosure weight is then determined by multiplying the weight per bolt by the total number of enclosure mounting bolts. Comparison with the actual weight of the enclosure and its components then indicates whether or not the mounting bolt stresses are exceeded. If the comparison shows that the maximum safe weight per bolt is exceeded, steps should be taken to increase the effective bolt area by welding the enclosure to its mounting, increasing the number of mounting bolts, adding top braces to a wall, or using another appropriate method to ensure safe operation during seismic disturbance.

TABLE A3.10-1
STANDARD ENCLOSURES

Curve	Enclosure	Width	Depth	Mode of Failure
C1	Instrument Rack	24"	24"	
	Instrument Rack	24"	30"	
	Vertical Board	24"	24"	Side to Side
	Vertical Board	24"	30"	
	Benchboard	24"	48"	
	Benchhoard	24"	54"	
C2	Instrument Rack	30"	30"	
	Instrument Rack	30"	24"	
	Instrument Rack	48"	24"	
	Instrument Rack	60"	24"	Front to Back
	Instrument Rack	72"	24"	OR
	Instrument Rack	96"	24"	Back to Front
	Vertical Board	36"	24"	
	Vertical Board	48"	24"	
	Vertical Board	60"	24"	
	Vertical Board	72"	24"	
	Vertical Board	96"	24"	
C3	Instrument Rack	48"	30"	
	Instrument Rack	60"	30"	
	Instrument Rack	72"	30"	
	Instrument Rack	96"	30"	
	Vertical Board	36"	30"	Front to Back
	Vertical Board	48"	30"	OR
	Vertical Board	60"	30"	Back to Front
	Vertical Board	72"	30"	
	Vertical Board	96"	30"	
C4	Console	96"	42"	Back to Front
C5	Benchboard	48"	54"	
				Side to Side
	Benchboard	48"	48"	
C6	Benchboard	72"	48"	
	Benchboard	96"	48"	Front to Back
	Console	96"	48"	
C7	Benchboard	72"	54"	
				Back to Front
	Benchboard	96"	54"	

PART II

Part II presents the necessary assumptions and equations for the calculation of the maximum normal and shear stresses in the mounting bolts of any enclosure under seismic disturbance. The following assumptions and conventions are made:

a. The enclosure under consideration is assumed to be a rigid body in equilibrium with respect to its mounting.

b. The forces on the enclosure due to seismic accelerations are assumed to act through the enclosure's center of gravity.

c. The enclosure is assumed to have a known weight W as well as a known center of gravity located at X, Y, Z with respect to a right-handed coordinate system.

d. The right-handed coordinate system is arbitrarily assumed to be located at the front left-hand lower corner of the enclosure with the positive X-axis to the right along the front edge, the positive Y-axis toward the back of the enclosure, and the positive Z-axis toward the top of the enclosure.

e. The stresses on the enclosure mounting bolts are assumed to be greatest when the horizontal component of the floor acceleration is perpendicular to a side of the enclosure and the vertical component of the acceleration is downward.

f. It is assumed that the enclosure tends to rotate about an axis parallel to either the X-axis or the Y-axis, dependent upon the direction of the horizontal acceleration. The location of the axis of rotation is dependent upon the mounting configuration of the enclosure.

g. There is assumed to be no friction between the enclosure and its mounting.

h. The horizontal shear force due to the horizontal component of the acceleration is assumed to be distributed equally among the mounting bolts.

i. All mounting bolts are assumed to be identical.

The following procedure outlines the equations involved in determining the mounting bolt stresses.

From the geometric configuration of the mounting bolts it is found that the tension forces in the bolts are related by

$$F_i = \frac{d_i}{d_j} F_j, \qquad (1)$$

where Fi and Fj are the tension forces acting on the i-th and the j-th bolts, respectively, and di and dj are the perpendicular distances of the i-th and the j-th bolts, respectively, from the axis about which the enclosure tends to rotate. When the enclosure is mounted directly to the floor, the axis of rotation will be an edge of the enclosure. For other mounting configuration, care must be exercised in determining the axis.

Summing moments about the enclosure's axis of rotation, the equation relating the unknown bolt tension forces to known quantities is found to be

$$F_1 d_1 + F_2 d_2 + \ldots + F_N d_N = W[A1 \bullet Z + (A2 - 1)L], \tag{2}$$

where N is the number of mounting bolts, A1 and A2 are the relative magnitudes of the horizontal and vertical components of the floor acceleration, respectively, and L is the perpendicular distance between the line of action of the vertical acceleration through the center of gravity and the axis about which the enclosure tends to rotate.

Substituting (1) into (2), the j-th tension force is

$$F_j = \frac{d_j \bullet W\,[\ A1 \bullet Z + (A2 - 1)L]}{d^2{}_2 + d^2{}_2\ +\ .\ .\ .\ \ + d^2{}_N} \tag{3}$$

The other tension forces are determined using Equation (1).

The tension stress T is related to the tension force by

$$T = \frac{F}{A} \tag{4}$$

Where A is the effective cross-sectional area of a mounting bolt.

Summing forces in the direction of the horizontal force acting upon the enclosure and making use of assumptions 7 and 8, the shear stress on the i-th bolt is

$$S = \frac{W \bullet A1}{N \bullet A} \tag{5}$$

Due to the combined tension and shear stresses, the maximum tension stress, (Ti) , and the maximum shear stress, (S) present in the i-th bolt are

$$T_i \max = \frac{T}{2} i\ +\ \frac{T}{2} i^2\ \ + S_i{}^2 \tag{6}$$

and

$$S_i \max = \sqrt{\frac{T}{2} i^2 + (S_i)^2} \tag{7}$$

For a detailed derivation of Equations (6) and (7), the reader is directed to Strength of Materials, by Ferdinand L. Singer, Chapter 9, Section 6.

To apply the above equations to determine the maximum tension and shear stresses, the following is required:

Center of Gravity	X, Y, Z Inches
Horizontal Seismic Acceleration	A1 - G
Vertical Seismic Acceleration	A2 - G

Number of Bolts	N
Area Each Bolt	A Square Inches
Bolts distance from Axis of Rotation	d_1 , $d_2 ... d_N$ Inches

PROCEDURE:

a. Determine the axis about which the cabinet tends to rotate for a given floor motion.

b. Determine, using Equation (3), the tension force acting on the j-th mounting bolts (arbitrarily choose one).

c. Determine the tension forces acting on the remaining mounting bolts from application of Equation (1).

d. Calculate the tension stress acting on each bolt using Equation (4) and the results of Step 3.

e. Calculate the horizontal shear stress from Equation (5).

f. Determine the maximum tension stresses using Equation (6) and the results of Steps 4 and 5.

g. Determine the maximum shear stresses using Equation (7) and the results of Steps 4 and 5.

h. Compare these maximum stresses and allowable stresses of one half the ultimate strength (in PSI) for the bolt material.

PART III

I. PURPOSE

The purpose of Part III is to document a static seismic analysis which was performed to verify that the mounting bolts of the standard cabinets are capable of withstanding seismic environment.

II. SCOPE

The scope of this report is limited to the static analysis of the mounting bolt stresses of five (5) standard cabinets. The standard cabinets are:

a. Area Radiation Monitor, 236x400 (911)

b. TIP Control, 236x401 (913)

c. Start-up Neutron Monitor, 236x402 (936)

d. Power Range Monitor, 236x403 (937)

e. Rod Position Information System, 236x404 (927)

III. DISCUSSION

The Seismic Design Guide, 225A4582, was used in conducting the static seismic analysis. Each cabinet was assumed to be floor mounted using 5/8" bolts in all mounting holes. The maximum safe tension stress and maximum safe shear stress was assumed to have a horizontal acceleration of 1.5G and a vertical [1] acceleration of 0.5G. The weight of each cabinet was estimated using the weight of each major component listed in the parts lists for each cabinet. The height of the center of gravity of each cabinet was calculated using the weight and center of gravity of each of the major components.

The following data sheets include the necessary information for determining the factor of safety for each cabinet.

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[1] Equal to one-half the ultimate strength as given in Machinery's Handbook, Fourteenth Edition.

SEISMIC DESIGN VERIFICATION DATA SHEET

Cabinet Name, Area Radiation Monitor (MPLH12P605)

Applied Horizontal Acceleration	1.5 G
Applied Vertical Acceleration	0.5 G
Tension Stress (Maximum Safe)	28,000 PSI
Weight of Cabinet	675 Lbs.
Number of Mounting Bolts	4
Height of Center of Gravity	48 Inches
Maximum Allowable Weight Per Bolts (From Curve No. CI on Page 8 of Seismic Design Guide, 225A4582)	830 Lbs/Bolt
Maximum Allowable Cabinet Weight 830 Lbs/Bolt * 4 Bolts =	3,320 Lbs.

$$\text{Factor of Safety} = \frac{\text{Maximum Allowable Weight}}{\text{Weight}} = 4.9$$

Cabinet Names: TIP Control, (H12P607)

Applied Horizontal Acceleration	1.5 G
Applied Vertical Acceleration	0.5 G
Tension Stress (Maximum Safe)	28,000 PSI
Shear Stress (Maximum Safe)	21,000 PSI
Weight of Cabinet	755 Lbs.
Number of Mounting Bolts	8
Height of Center of Gravity	50 Inches
Maximum Allowable Weight Per Bolt (From Curve No. C3 on Page 8 of Seismic Design Guide, 225A4582)	1,110 Lbs.
Maximum Allowable Cabinet Weight 1,110 Lbs/Bolt * 8 Bolts =	8,880 Lbs.

$$\text{Factor of Safety} = \frac{\text{Maximum Allowable Weight}}{\text{Weight}} =11.7$$

Cabinet Name: Start-Up Neutron Monitor, (H12P633)

Applied Horizontal Acceleration	1.5 G
Applied Vertical Acceleration	0.5 G
Tension Stress (Maximum Safe)	28,000 PSI
Shear Stress (Maximum Safe)	21,000 PSI
Weight of Cabinet	1,910 Lbs.
Number of Mounting Bolts	12
Height of Center of Gravity	50 Inches
Maximum Allowable Weight Per Bolt (From Curve No. C3 on Page 8 of Seismic Design Guide, 225A4582)	1,110 Lbs/Bolt

SEISMIC DESIGN VERIFICATION DATA SHEET (Continued)

Maximum Allowable Cabinet Weight	
1,110 Lbs/Bolt * 12 Bolts =	13,320 Lbs

Factor of Safety = Maximum Allowable Weight / Weight = 11.9

Cabinet Name: Power Range Monitor, (328x105) (H12P608)

Applied Horizontal Acceleration	1.5 G
Applied Vertical Acceleration	0.5 G
Tension Stress (Maximum Safe)	28,000 PSI
Shear Stress (Maximum Safe)	21,000 PSI
Weight of Cabinet	4,345 Lbs.
Number of Mounting Bolts	40
Height of Center of Gravity	46 Inches
Maximum Allowable Weight Per Bolt (From Curve No. C3 on Page 8 of Seismic Design Guide, 225A4582)	1,210 Lbs/Bolt
Maximum Allowable Cabinet Weight	
1,210 Lbs/Bolt * 40 Bolts =	48,400 Lbs.

Factor of Safety = Maximum Allowable Weight / Weight 11.1

Cabinet Name: Rod Position Information System, (H12P615)

Applied Horizontal Acceleration	1.5 G
Applied Vertical Acceleration	0.5 G
Tension Stress (Maximum Safe)	28,000 PSI
Shear Stress (Maximum Safe)	21,000 PSI
Weight of Cabinet	2,500 Lbs.
Number of Mounting Bolts	20
Height of Center of Gravity	45 Inches
Maximum Allowable Weight Per Bolt (From Curve No. C3 on Page 8 of Seismic Design Guide, 225A4582)	1,225 Lbs/Bolt
Maximum Allowable Cabinet Weight	
1,225 Lbs/Bolt * 20 Bolts =	24,500 Lbs.

Factor of Safety = Maximum Allowable Weight / Weight 9.8

IV. CONCLUSION

Review of the Factor of Safety of each standard cabinet indicates that the mounting bolts of each cabinet are capable of withstanding seismic disturbance as specified in the Seismic Design Guide.

ATTACHMENT B3.10
SAMPLE PANEL FREQUENCY ANALYSIS

The method of analysis used to determine the resonant frequency of the panel is as follows:

a. Calculate the moment of inertia of the corner post structure.

b. First assume a simplified structure and calculate the frequency using the expression:

$$f = (1/2\pi\sqrt{kg/w} = ((\sqrt{g})/2\pi)\sqrt{k/w} = 3.13/\sqrt{w/k}$$

$$f = 3.13/\sqrt{\delta}$$

Where:

f = frequency
g = 386 in./sec^2
k = spring rate #/in.
w = weight #
δ = deflection = w/k

weight distribution is assumed to be uniform.

c. Additional structural components are added and the moment and frequency recalculated.

The calculated resonant frequency of 7.4 Hz for the panel and 5.9 Hz for the benchboard was obtained using only the corner posts and the top. The addition of skin (3/8-in. steel) and 2-in. x 1/4-in. steel stiffeners will raise the frequency further. This proves that resonances cannot exist in the unstable region below 5 Hertz.

FIRST APPROXIMATION

For First Approximation lump the 4 corner posts together and assume the panel is a cantilever beam fixed on one end and uniformly loaded (see Figure B3.10-1).

The natural frequency is 2.6 Hz so we will have to use more of the structure.

SECOND APPROXIMATION

For a second approximation, consider two 0.18" x 30" barriers in addition to the corner posts. The plan view of the panel is shown in Figure B3.10-2.

In the X direction just one barrier will raise the frequency to 30 Hz. Use 4 inches of the back panel for each of the two barriers (see Figure B3.10-3) and the natural frequency in the Y direction becomes 4 Hz.

The deflection equation used so far is very conservative: it assumes that the 4 corner posts are lumped together and that the structure can deflect like a simple cantilever beam. Actually the corners are separated by an angle frame which is stiffer than the corner posts. This will force the structure to deflect as shown in Figure B3.10-4.

In the simulated model we are not conservative (if we used all of the members) but we are very close. The reason we are not quite correct is because the stiff top frame will deflect slightly as shown below. The calculated frequency is 7.4 Hz which is above the necessary 5 Hz.

The benchboard H11 P601 which weighs 4000 pounds, the calculated natural frequency is 5.9 Hz which is still above the 5 Hz test frequency minimum.

NOTE: This neglects the barriers, the end and front panels, top plate, the stiffening of the lower part of the structure due to the bench board geometry, and all other members of the structure.

3.11 ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

3.11.1 Introduction

Environmental equipment qualification efforts for the Clinton Power Station Unit 1 began with the issuance of original equipment procurement specifications. These documents contained requirements to ensure General Design Criteria (10 CFR 50 Appendix A) 1, 2, 4, and 23 were satisfied and included IEEE 323 and 344.

Since the issuance of IE Bulletin 79-01B, NUREG-0588 (Reference 2) and the Commission Memorandum and Order (CLI-80-21) of May 23, 1980 (Reference 1), an effort was initiated to compare the Clinton Environmental Qualification (CEQ) program against the requirements as stated in these documents. This included recalculation or verification of environmental parameters (radiation, temperature, pressure, humidity) to ensure consistency with guidelines contained in NUREG-0588, Rev. 1, Category 1 requirements.

Also, a complete electrical systems analysis was performed to revalidate the listing of electrical equipment and components required to satisfy six safety goals for plant operation and shutdown. These are:

a. Safe shutdown

b. Containment isolation

c. Core coverage

d. Residual heat removal

e. Containment integrity

f. Effluent control

The objective of this study was to establish a comprehensive list of Class IE electrical equipment in harsh environmental zones of the plant that require qualification to support these safety goals.

The environmental qualification program consists of essentially three phases:

a. The assessment and evaluation phase that was performed for equipment already qualified to specified environmental conditions prior to issuance of NUREG-0588. Furthermore, this appraisal verifies the actual basis of qualification for the equipment against the NUREG-0588 criteria. This phase is discussed further in Subsection 3.11.5.

b. The ongoing qualification phase that uses the environmental parameters of Reference 16, which are based on NUREG-0588. This phase is discussed further in Subsection 3.11.6.

c. The requalification phase includes the following options:

1. Relocation

2. Reanalysis

3. Retest

4. Replacement of equipment that does not adequately qualify to NUREG-0588 requirements with a qualified replacement.

This phase is discussed further in Subsection 3.11.7.

3.11.2 Definitions

a. Harsh environmental zone – An area in the plant that experiences environmental conditions resulting from a design basis event, such as loss of coolant accident (LOCA) or high energy line break (HELB) or main steam line break (MSLB). Also, refer to Subsection 3.11.9.2.1 on radiation.

b. Mild environmental zone – An area in the plant that experiences an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences (extremes/abnormal). As an example, the loss of offsite electrical power (LOOP) could result in loss of ventilating equipment and change the normal to abnormal conditions but could still be considered a mild environment. Here, the seismic event is the only design basis event of consequence. Normal plant operation, in general, includes power operation, start-up, shutdown, and refueling conditions. Also, refer to Subsection 3.11.10.2 on radiation.

c. Engineered safety feature (ESF) systems - These systems are provided to mitigate the consequences of design basis accidents and are discussed in detail in Chapter 6.

d. High energy lines - Those lines at pressures above 275 psig and/or temperature above 200° F. Refer to Subsection 3.6.1.1.1.b.

3.11.3 Safety Systems and Supporting Equipment

In the physical layout of the CPS plant equipment, specific attention was given to the location of Class IE equipment in the containment building. As far as practical, redundant safety-related cooling is provided for essential control panels, auxiliary equipment panels, cable spreading areas, and essential switchgear. Furthermore, all Class IE electrical system logics, interlocks, controllers, indicators, recorders, relays, etc. that comprise the control circuitry are located in the mild environmental zones wherever feasible.

The engineered safety feature systems are the first line systems required to achieve or maintain safe reactor shutdown, containment isolation, reactor core cooling, containment heat removal, core residual heat removal, and prevention of significant release of radioactive materials to the environment. The functional aspects of the systems that are required to support these safety functions are described separately in the appropriate sections of the USAR for each system and support equipment.

All Class 1E electrical equipment located in a harsh environment, as defined in the scope of 10 CFR 50.49 (Reference 12), is included in List 1 of Nuclear Station Engineering Standard MS-02.00, Maintenance of Equipment Qualification Program Manual (Reference 13).

3.11.4 NUREG-0588 Parameters Considered in Qualification Phase

a. Temperature - This parameter for each plant environmental zone is discussed in Subsection 3.11.9, and the discrete values for each zone are provided in Reference 16.

b. Radiation - This parameter for each plant environmental zone is discussed in Subsection 3.11.9, and the discrete values for each zone are provided in Reference 16.

c. Aging - Aging effects on equipment are considered in the qualification program. This includes electrical and mechanical cycling for equipment wherever appropriate. For most equipment, the Arrhenius methodology is used for determining the accelerated thermal aging requirements, and the aging acceleration rate is defined for each component in the specific environmental qualification program. Aging is addressed in greater detail in specific environmental qualification (EQ) binders prepared for the equipment and available in the Clinton Power Station Central File.

d. Seismic and dynamic - For IE electrical equipment, the dynamic qualification is discussed in Section 3.10.

e. Chemical environment - Since demineralized water is used for all safety systems, chemical spray is not a concern for Clinton Environmental Qualification.

f. Pressure - This parameter for each plant environmental zone is discussed in Subsection 3.11.9, and the discrete values for each zone are provided in Reference 16.

g. Humidity - This parameter for each plant environmental zone is discussed in Subsection 3.11.9, and the discrete values for each zone are provided in Reference 16.

h. Submergence - This parameter for each plant environmental zone is discussed in Subsection 3.11.9, and the discrete values for components are addressed in the applicable EQ binders.

i. Synergism - This effect has been addressed in the EQ binders for those materials affected by this phenomenon.

j. Dust - There are general administrative housekeeping procedures to maintain plant cleanliness at acceptable levels, including a maintenance and replacement schedule for HVAC filter units.

k. Margins - For equipment that is type tested, appropriate margins have been applied to the service conditions in conformance with Subsection 6.3.1.5 of IEEE 323 (Reference 3). This is addressed in more detail in the EQ binders. A one-hour margin has been applied when operability times are less than 10 hours. In cases where a deviation from this is taken for specific equipment, a justification is provided in the applicable EQ binders.

l. Test sequence - For equipment that is type tested, the sequence of testing is in conformance with Subsection 6.3.2 of Reference 3 unless otherwise noted and justified in the detailed EQ binders.

m. Periodic surveillance and maintenance - Qualified life is established based on the qualification of the most limiting material identified in the equipment. A maintenance and surveillance schedule is established for the equipment. This is addressed further in Subsection 3.11.11 and in the individual qualification packages.

n. Containment spray - This parameter for environmental zones which are subject to containment spray is discussed in Subsection 3.11.9.2.2 and is addressed further in the individual EQ binders as applicable. The operation of the containment spray system is addressed in Subsection 6.2.2. Environmental Zones H-1, H-15, H-26, H-37, and H-44 are subject to containment spray.

3.11.5 Assessment and Evaluation Phase

The assessment phase includes an identification of all equipment important to safety in terms of its function during normal and abnormal environments. If the equipment is used in several systems throughout the plant, the most demanding safety function and the most severe environment is the condition used to assess its qualification. Each piece of BOP and NSSS IE electrical equipment is identified specifically by manufacturer, model number and type. Additionally, for all active equipment, the time required to perform its respective essential safety function is identified.

In this phase of the work, the plant areas are divided into environmental zones that are determined by various plant events. The zone classifications are harsh and mild environmental zones, and are described in Subsection 3.11.9. The applicable normal, abnormal, and accident plant conditions for each harsh zone are shown in Reference 16. The assessment and evaluation are based on the parameters identified or referenced in Reference 16.

Further, in this phase a detailed appraisal of equipment's capability to perform its safety function in an accident environment associated with LOCA or HELB was performed based on available qualification test reports.

3.11.5.1 BOP Assessment and Evaluation Phase

For each unique piece of BOP Class IE electrical equipment located within a harsh environmental zone, an equipment qualification report was submitted by the equipment vendor (or testing laboratory) in accordance with the requirements of the equipment technical specifications.

The assessment and evaluation of this equipment is based on the environmental parameters listed in Reference 16.

All BOP Class IE equipment is included in List 1 of MS-02.00. This list also provides equipment number, manufacturer, type/model number and EQ and/or SQ binder numbers.

For all environmentally qualified electrical equipment listed in List 1 a detailed EQ checklist (which forms a part of the EQ binder) in compliance with NUREG-0588 is prepared and sent to the Clinton Power Station Central File (see Subsection 3.11.12).

3.11.5.2 NSSS Assessment and Evaluation Phase

All NSSS Class IE equipment is included in List 1 of MS-02.00. This list also provides equipment number, manufacturer, model number and EQ and/or SQ binder numbers.

For each unique piece of NSSS Class IE electrical equipment located within a harsh zone, an equipment qualification report was submitted by the equipment vendor (or testing laboratory) in accordance with the requirements of the equipment technical specifications.

The assessment and evaluation of this equipment is based on the environmental parameters listed in Reference 16.

For all environmentally qualified electrical equipment listed in List 1, a detailed EQ checklist (which forms a part of the EQ binder) in compliance with NUREG-0588 is prepared and sent to the Clinton Power Station Central File (see Subsection 3.11.12).

3.11.6 Status of Ongoing Qualification Efforts

The ongoing qualification phase uses the environmental zone parameters of Reference 16 for both BOP and NSSS equipment qualification.

3.11.7 Requalification Phase

3.11.7.1 Equipment in Harsh Environments

Instances in which Class IE equipment did not meet the requirements of NUREG-0588 were resolved by implementing one of the following options:

a. Relocation of the equipment from a "harsh" to a less severe "harsh" or a "mild" environment provided the system logic or other design functions were not affected.

b. Replacement of the equipment with upgraded equipment.

c. Protection of the equipment such that a more realistic set of environmental conditions could be achieved for the specific location.

d. Retesting the equipment using appropriate environmental parameters based on its unique location.

3.11.7.2 Equipment in Mild Environments

Since equipment in mild environments is not exposed to severe environmental conditions of a design basis event, requalification to levels described for harsh environment equipment is not required per Reference 12.

3.11.8 Electrical Equipment Tabulation and Format

List 1 of MS-02.00 includes the Class IE equipment for all environmental zones (harsh and mild) in alphabetical order. The information presented in this list for electrical equipment 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) and the generic name (type) of the equipment.

b. Equipment manufacturer: Identifies the manufacturer of the equipment.

c. Type/Model number: Provides the equipment type/model or catalog number.

d. EQ binder number: Provides a reference to the environmental qualification binder for the equipment in harsh environment.

e. SQ binder number: Provides a reference to seismic qualification binder for seismically qualified equipment in harsh and mild environments.

3.11.9 Plant Environmental Zones

The plant areas containing Class IE equipment are divided into two zones based on the environmental conditions that are expected to occur as a result of various plant events. These zone classifications are termed harsh and mild environmental zones (see Section 3.11.2a and Section 3.11.2b). The mild environmental zone is discussed in Subsection 3.11.10. The harsh environmental zones are discussed in the subsections below.

The environmental parameters for the abnormal and normal service conditions represent conservative selections chosen to bound the real conditions that may occur in these zones. A more refined or more detailed analysis has been performed for specific equipment to establish more realistic and representative environmental parameters than the values specified for the specific environmental zone. In those cases where unique calculations are prepared, they have been made part of the environmental qualification records.

3.11.9.1 Harsh Environmental Zones Due to LOCA, HELB or LOOP

The development of the environmental conditions within the harsh environmental zones is based on results of analyses of postulated accidents. The postulated accidents considered are loss-of-coolant (LOCA), high energy line break (HELB), and a loss-of-offsite power (LOOP).

The following sections summarize the basis for the establishment of the harsh environmental zones. LOCA, HELB and LOOP were investigated and the bounding conditions presented. Where possible, several plant areas have been grouped into a single zone with an H-x designation that bounds the environmental conditions in each of the individual areas. The various environmental zones are represented on plant general arrangement drawings as environmental Zone Maps (Drawing M01-1600 Sheets 6 through 21).

The environmental conditions for each zone pertaining to pressure, temperature, relative humidity, duration and submergence are addressed in Subsections 3.11.9.3 through 3.11.9.62 and Reference 16.

The "Submergence or Spray" section applies to the containment and drywell for design considerations resulting from suppression pool dynamic events such as suppression pool swell or weir swell, and from containment spray. The radiation environment is discussed in Subsection 3.11.9.2.1 and Reference 16.

Flood protection for the Clinton Power Station is discussed in Subsection D3.6.4.

3.11.9.2 Radiation and Containment Spray

3.11.9.2.1 Radiation

The design basis accidents addressed in the determination of the radiation environment are the loss of coolant accident (LOCA), the fuel handling accident (FHA) and the high energy line break accident (HELB). The LOCA produces the most severe radiation environment, and as such is used as the design basis accident where it is applicable. The environmental conditions produced by the HELB accident last for a relatively short time, such that the associated radiation environment, which is expressed in terms of total integrated radiation dose, is not significant. The FHA also produces less severe radiation environment compared to LOCA, but is used as the design basis accident for those components which are required to survive a FHA, but not the LOCA.

The radiation environment after a design-basis accident is determined based upon the assumptions provided in NUREG-0588. The source terms are calculated using the reactor data provided in Table 12.2-1 and the RACER Code (Reference 2 of Subsection 12.3.5). It is assumed that the containment leaks into the gas-control boundary at the design-basis leak rate, as specified in Subsection 6.2.6, and that the gas-control boundary is evacuated by the standby gas treatment system (SGTS). The exhaust rate and filter efficiencies of the SGTS are listed in Subsection 6.5.1. The effects of post LOCA recirculation fluids have been included in the determination of radiation environment in accordance with NUREG-0737, Section II.B.2.

A harsh environment attributed to radiation is defined as an environment with a total integrated dose value greater than 1×10^4 rad (C), in which the major dose contribution is from a postaccident condition.

The Radiation Qualification Dose values to be utilized for testing of equipment in each environmental zone are listed in Reference 16. The dose values are the sum of gamma radiation dose values and beta radiation dose values. The conservatively calculated total dose value for each zone is less than, or equal to, the value specified as the radiation qualification dose. The calculated dose is the sum of the integrated dose for 40 years of normal operation plus one year of radiation exposure in a postaccident condition.

3.11.9.2.2 Containment Spray

Equipment which is located in the containment and outside of the drywell, except equipment located in cubicles, will be environmentally qualified for the containment spray requirements. The following spray test requirements are used to simulate the containment spray system (Reference 17):

Initiation of spray	-	10 minutes after LOCA
Spray duration	-	one hour
Spray loading	-	one gpm per square foot
Water chemistry	-	demineralized water

3.11.9.3 Environmental Zone H-1

Zone H-1 is the suppression pool in the containment building. This zone is identified as area C.1.1 in Drawing M01-1600-6. For this zone, the bounding environmental conditions result from postulated design-basis LOCA events or from the normal and abnormal operation of the main steam safety/relief valves. The conditions arising from safety/relief valve operation are not discussed here. For a full discussion of these phenomena, including the short-term LOCA pool swell, condensation oscillation, and chugging, refer to Sections A3.8 and A3.9.

a. Pressure

The determination of the design pressure is discussed in Section 6.2. The pressure range is given in Reference 16.

b. Temperature

The temperature profile is given in Figures 6.2-3 (Curve 2) or 6.2-12 (Curve 2) and 6.2-7a (Curve 3) and 6.2-7b (Curve 3).

The determination of the design temperature is discussed in Section 6.2.

c. Relative humidity

Since this zone is normally flooded, the relative humidity for this zone cannot be defined.

d. Duration

The time histories for the spectrum of LOCA events are given in Section 6.2 and Figures 6.2-2 (wetwell), 6.2-3 (Curve 2), 6.2-6a (Curve 2), 6.2-6b (Curve 2), 6.2-7a (Curve 3), 6.2-7b (Curve 3), 6.2-11 (wetwell), and 6.2-12 (Curve 2). The temperature and pressure at the end of these curves is conservatively assumed to persist up to 100 days.

e. Submergence or Spray

This zone is flooded under normal and accident conditions.

3.11.9.4 Environmental Zone H-2

Zone H-2 is the lower elevation of the drywell. This zone is denoted as area C.1.2 in Drawing M01-1600-6. The bounding environmental conditions in this zone result from design-basis LOCA events.

a. Pressure

The pressure environment for this zone is defined as the design basis pressure transient for the drywell. The determination of this design basis is discussed in Section 6.2. Table T1 of Reference 16 shows the envelope of the LOCA pressure conditions addressed in Section 6.2.

b. Temperature

The temperature environment for this zone is defined as the design basis temperature transient for the drywell. The determination of the design basis is discussed in Section 6.2, Table T1 of Reference 16 shows the envelope of the LOCA temperature conditions addressed in Section 6.2.

c. Relative humidity

Table T1 of Reference 16 gives the relative humidity envelope as a function of time for the environmental qualification of equipment. The environment is conservatively assumed to be all steam for the first 6 hours following an accident and at 100% relative humidity thereafter.

d. Duration

The time histories for the spectrum of LOCA events are shown in Section 6.2 and Figures 6.2-2 (drywell), 6.2-3 (Curve 1), 6.2-6a (Curve 1), 6.2-6b (Curve 1), 6.2-7a (Curve 1), 6.2-7b (Curve 1), 6.2-11 (drywell), and 6.2-12 (Curve 1). The time history given in Table T1 of Reference 16 envelops all these events plus the small HELB.

e. Submergence or Spray

This zone is normally dry and is provided with sumps and drains to maintain this condition during normal operation. Following a design basis LOCA, the drywell depressurization will cause water to be drawn into the drywell through the LOCA vents. The nature of this transient is such that the drywell could be flooded to the top of the weir wall (see Reference 10).

3.11.9.5 Environmental Zone H-3

Zone H-3 is the portion of the drywell inside the pedestal under the RPV. This zone is designated area C.1.3 in Drawing M01-1600-6. The bounding environmental conditions for this zone result from design-basis LOCA events. For abnormal conditions during scram refer to Figure T37 of Reference 16.

a. Pressure

The pressure environment for this zone is defined by the spectrum of LOCA events discussed in Section 6.2. The large HELB and the small HELB are, in this environmental zone, a subset of the spectrum of LOCA events. The bounding pressure transient in this environmental zone is a composite of the large HELB and small HELB pressure transients given in Table T1 of Reference 16.

b. Temperature

The temperature environment for this zone is defined by the spectrum of LOCA events discussed in Section 6.2. The bounding temperature transient is given in Table T1 of Reference 16 as the small HELB conditions. The small HELB in Table T1 of Reference 16 is a subset of the spectrum of LOCA events and its temperature transient bounds that of all other line breaks in this environmental zone.

c. Relative humidity

The relative humidity for this zone is conservatively assumed to be 100% except for the time period immediately following the accident when the atmosphere is considered to be all steam. See Table T1 of Reference 16.

d. Duration

The time histories for the spectrum of LOCA events are enveloped by the time histories presented in Table T1 of Reference 16. These time histories are the results of GE analysis per Reference 7 and enveloped per Reference 8.

e. Submergence or Spray

Same as Section e for Zone H-2.

3.11.9.6 Environmental Zone H-4

Zone H-4 is an area of the fuel building. This zone is identified as area F.2.1 in Drawing M01-1600-7. The bounding environmental conditions for this zone result from a LOCA.

a. Pressure

The bounding pressure environment for this zone is defined by the design parameters for the standby gas treatment system and the secondary containment. The accident condition analysis is discussed in Subsection 6.2.3 and Table T7 of Reference 16 gives the envelope for the pressure conditions following an accident.

b. Temperature

The bounding temperature environment for this zone is defined by the design parameters for the standby gas treatment system and the secondary containment. The accident condition analysis is discussed in Subsection 6.2.3 and Table T7 of Reference 16 gives the envelope for the temperature conditions following an accident.

c. Relative humidity

The relative humidity following a LOCA does not exceed 96% is given in Reference 16.

d. Duration

The envelope of the transient events is given in Table T7 of Reference 16. This envelope is developed from the analysis discussed in Subsection 6.2.3.

3.11.9.7 Environmental Zone H-5

This zone is composed of the majority of the fuel building. The zone is identified as areas F.1.1, F.1.2, F.1.3, F.1.4, F.1.5, F.1.6, F.1.9, F.1.11, F.2.2, F.2.3, F.2.4, F.2.5, F.2.6, F.2.7, and F.3.1 in Drawing M01-1600 Sheets 6 through 9. The bounding environmental conditions result from a LOCA.

a. Pressure

Same as Zone H-4.

b. Temperature

Same as Zone H-4.

c. Relative humidity

Same as Zone H-4.

d. Duration

Same as Zone H-4.

3.11.9.8 Environmental Zone H-6

Zone H-6 is the HPCS pump cubicle in the fuel building. This zone is denoted as area F.1.7 in Drawing M01-1600-6. For equipment qualification purposes environmental conditions are based on the bounding environmental conditions as discussed below.

a. Pressure

The pressure environment is defined by Reference 7 which presents a Mark III generic analysis of the accident transients and the envelope of the analysis of Reference 8. This envelope bounds the results of analysis discussed in Subsection 6.2.3. Reference 16 presents the envelope for the LOCA events and the HELB accidents.

b. Temperature

The temperature environment is defined by the Mark III generic analysis performed by GE (Reference 7) and the envelope of the analysis of Reference 8. Table T2 of Reference 16 presents the envelope for the LOCA events and Table T3 of Reference 16 presents the envelope for HELB accidents.

c. Relative humidity

Table T2 of Reference 16 shows the relative humidity envelope determined by the generic Mark III analysis. Table T3 of Reference 16 shows the relative humidity envelope for the postulated HELB events.

d. Duration

The time histories for the LOCA and HELB events are given in Tables T2 of Reference 16 and T3 of Reference 16.

3.11.9.9 Environmental Zone H-7

Zone H-7 is composed of the fuel building floor drain tank cubicles, the fuel building tunnel, and the fuel pool heat exchanger cubicle. These areas are identified as F.1.8, F.1.10, F.1.12, and F.2.8 in Drawing M01-1600 Sheets 6 and 7. The bounding environmental conditions in these areas result from a LOCA.

a. Pressure

Same as Zone H-4.

b. Temperature

Same as Zone H-4.

c. Relative humidity

Same as Zone H-4.

d. Duration

Same as Zone H-4.

3.11.9.10 Environmental Zone H-8

Zone H-8 is composed of general areas of the auxiliary building. These areas are identified as A.1.11, A.1.13 and A.1.14 in Drawing M01-1600-6. The bounding environmental conditions for this area result from a LOOP.

a. Pressure

The areas in this zone are outside the secondary containment and thus are not subject to any pressure effects due to a LOCA. The pressure variations which would occur in this area during a LOOP transient are given in Table T10 of Reference 16.

b. Temperature

The areas in this zone are outside the secondary containment and thus are not subject to the temperature effects due to a LOCA. The bounding temperature is determined as the result of a postulated LOOP. The extreme expected temperatures are cited in Table T10 of Reference 16.

c. Relative humidity

The extreme values of relative humidity for this area are given in Table T10 of Reference 16.

d. Duration

The only transients of significance relate to the temperature. The transient is assumed to last for the duration of the LOOP.

3.11.9.11 Environmental Zone H-9

Zone H-9 is composed of the access aisle and floor drain pump cubicle denoted areas A.1.1 and A.1.2 in Drawing M01-1600-6. The bounding environmental conditions for this area result from a HELB in the pipe chase or a LOCA.

a. Pressure

The pressure conditions resulting from a LOCA are mitigated by the standby gas treatment system and held below 1 inch of water as discussed in Subsection 6.2.3.

The pressure conditions resulting from a HELB (in this area or in adjacent areas) or LOCA are given in Table T3 of Reference 16.

b. Temperature

The temperature conditions resulting from a LOCA are determined in Subsection 6.2.3 and the values are given in Figures T34 (Curve 2) and T36 (Curve 8) of Reference 16.

The temperature conditions resulting from a HELB in these areas or in adjacent areas are tabulated in Table T3 of Reference 16.

c. Relative humidity

The relative humidity for this zone is given in Table T3 of Reference 16 for the spectrum of breaks.

d. Duration

The LOCA pressure transient is inconsequential. The LOCA temperature transient is given as the "ECCS equipment rooms" and the "RHR-C pump room" shown in Figures T34 (Curve 2) and T36 (Curve 8) of Reference 16, respectively. The temperature at the end of the curve shown in Figure T36 is conservatively assumed to persist up to 100 days.

The HELB pressure conditions are of minor importance. The peak values given in Table T3 of Reference 16 are assumed to persist for the duration of the accident.

3.11.9.12 Environmental Zone H-10

Zone H-10 is the floor drain tank cubicle denoted as area A.1.3 in Drawing M01-1600-6. The bounding environmental conditions in this zone result from a HELB in the adjacent pipe chase or a LOCA.

a. Pressure

Same as Zone H-9.

b. Temperature

Same as Zone H-9.

c. Relative humidity

Same as Zone H-9.

d. Duration

Same as Zone H-9.

3.11.9.13 Environmental Zone H-11

Zone H-11 is composed of the RHR-C and LPCS pump cubicles designated areas A.1.4 and A.1.10 in Drawing M01-1600-6. The bounding environmental conditions for this zone are determined by a LOCA or by a HELB in the individual cubicles.

a. Pressure

The pressure conditions from a LOCA are mitigated by the standby gas treatment system and held below 1 inch of water as discussed in Subsection 6.2.3.

The pressure transients following a HELB (in the adjacent cubicles) or LOCA are detailed in Table T3 of Reference 16.

b. Temperature

The temperature transients following HELB's in adjacent areas are given in Table T3 of Reference 16.

The temperature transient following a LOCA is determined in Subsection 6.2.3 and the values are given in Figures T34 (Curve 2) and T36 (Curve 4 and Curve 8) of Reference 16.

c. Relative humidity

The range of relative humidities expected following the various accidents is given in Table T3 of Reference 16.

d. Duration

The LOCA pressure transient is inconsequential. The LOCA temperature transient is given as the "ECCS equipment rooms" shown in Figure T34 (Curve 2) of Reference 16 and "RHR-C pump room" and "LPCS pump room" shown in Figure T36 (Curve 4 and Curve 8) of Reference 16. The temperature at the end of the curve shown in Figure T36 is conservatively assumed to persist for up to 100 days.

The pressure and temperature time-history of a HELB in the adjacent cubicles is given in Table T3 of Reference 16. The pressures cited are assumed to persist for the duration of the transient.

3.11.9.14 Environmental Zone H-12

Zone H-12 is composed of the RHR Pump and Heat Exchanger Room B; RCIC Pump Room; RHR Pump and Heat Exchanger Room A, shown as areas A.1.6, A.1.7 and A.1.9 respectively on Drawing M01-1600, Sheet 6. Areas A.1.6 and A.1.9 are also shown on Drawing M01-1600 Sheet 7 through Sheet 9. The bounding environmental parameters are determined by a LOCA or a HELB in the individual cubicles or the adjacent RHR-C pump cubicle.

a. Pressure

The pressure transients for the LOCA, HELB in an adjoining cubicle, and HELB in the adjacent RHRC pump cubicle are shown in Table T3 of Reference 16. The peak conditions for the break in the individual cubicles is determined by the analysis of Reference 9. The remainder of the pressure transients are from the analysis in References 7 and 8.

b. Temperature

The temperature transients following HELB's in adjacent areas are given in Table T3 of Reference 16.

The temperature transient following a LOCA is determined in Subsection 6.2.3 and the values are given in Table T3 of Reference 16.

c. Relative humidity

The relative humidity for the HELB in the individual cubicles is not considered, rather an all steam environment is specified for conservativeness.

Table T3 of Reference 16 presents the bounding relative humidity for the other transients discussed in Section a above.

d. Duration

The peak values determined for the HELB in the adjacent cubicles are specified in Table T3 of Reference 16 are conservatively assumed to persist for the duration of the accident. The time-histories for the remainder of the transients are also shown in Table T3 of Reference 16.

3.11.9.15 Environmental Zone H-13

Zone H-13 is the RCIC instrument panel room shown as area A.1.8 in Drawing M01-1600-6. The bounding environmental conditions for this area result from the LOCA, a HELB in the RCIC pump cubicle, or a HELB in the RHR-A, RHR-B or RHR-C pump cubicles.

a. Pressure

The pressure transients for the LOCA, HELB in the RCIC cubicles, HELB in an adjoining cubicle, and HELB in the adjacent RHR-C pump cubicles are shown in Table T3 of Reference 16. The peak conditions for the break in the individual cubicles are determined by the analysis of Reference 9. The remainder of the pressure transients are from the analysis in References 7 and 8.

b. Temperature

Same as Zone H-12 for area A.1.7.

c. Relative humidity

Same as Zone H-12.

d. Duration

Same as Zone H-12.

3.11.9.16 Environmental Zone H-14

Zone H-14 is comprised of stairways and general areas of the auxiliary building that are denoted as areas A.2.11, A.2.20, A.2.22, and A.2.23 in Drawing M01-1600-7. The bounding environmental conditions for this zone result from a simultaneous LOCA and LOOP.

a. Pressure

The areas under consideration in this zone are outside of the secondary gas control boundary and thus do not experience any pressure effects from a LOCA. Also since no high energy lines are located in these areas the pressure transient is inconsequential. (See Table T10 of Reference 16.)

b. Temperature

The peak temperatures in this area will occur following the simultaneous LOCA and LOOP. The peak calculated temperatures for these zones are listed in Table T10 of Reference 16.

c. Relative humidity

The range of relative humidity following a LOOP is given in Table T10 of Reference 16.

d. Duration

The pressure consequences of a LOCA or LOOP are inconsequential. LOCA temperature conditions do not have a direct effect on the areas in this zone. Post-LOCA boundary conditions are used in computing the SGTS and secondary containment area bounding environmental temperatures. The transient is assumed to last for the duration of the LOCA.

3.11.9.17 Environmental Zone H-15

Zone H-15 is the wetwell area above the suppression pool in the containment. This zone is identified as area C.2.1 in Drawing M01-1600-7. The bounding environmental conditions for this zone result from a spectrum of LOCA events consisting of a large HELB or a small HELB as described in Table T4 of Reference 16.

a. Pressure

The pressure environment following a LOCA is based on the GE Mark III generic accident analysis of Reference 7. The CPS-unique analyses are discussed in Section 6.2. The generic analyses as enveloped in Reference 8 bound the CPS-unique analysis. The results of the generic analyses are given in Table T4 of Reference 16.

b. Temperature

The temperature environment following a LOCA is based on the GE Mark III generic accident analysis of Reference 7. The CPS-unique analyses are discussed in Section 6.2. The generic analyses as enveloped in Reference 8 bound the CPS-unique analysis. The results of the generic analyses are given in Table T4 of Reference 16.

c. Relative humidity

The relative humidity environment following a LOCA is based on the GE Mark III generic accident analysis of Reference 7. The CPS-unique analyses are discussed in Section 6.2. The generic analyses as enveloped in Reference 8 bound the CPS unique analysis. The results of the generic analyses are given in Table T4 of Reference 16.

d. Duration

Table T4 of Reference 16 gives the time-histories following design basis LOCA events.

e. Submergence or Spray

Equipment in this area may be submerged or exposed to a spray from pool swell or operation or the containment spray.

The pool swell phenomenon is described in Section A3.8.

3.11.9.18 Environmental Zone H-16

Zone H-16 is the drywell. This zone is denoted as area C.2.2 in Drawing M01-1600-7. The bounding environmental conditions for this zone result from design-basis LOCA events.

a. Pressure

Same as Zone H-2.

b. Temperature

Same as Zone H-2.

c. Relative humidity

Same as Zone H-2.

d. Duration

Same as Zone H-2.

e. Submergence or Spray

The portion of the drywell below floor Elevation 740 feet 9 inches will be subject to the flooding and/or spray effects of weir swell. The zone of influence of the weir swell phenomena is shown in Reference 10.

3.11.9.19 Environmental Zone H-17

Zone H-17 is composed of auxiliary building access areas A.2.1, A.3.1, A.3.5, A.3.8, A.3.9, A.4.1, A.4.5, A.4.10, and A.4.11 in Drawing M01-1600 Sheets 7 through 9.. The environmental conditions for this zone result from a LOCA or a HELB in ECCS pump cubicles.

a. Pressure

The pressure transients resulting from a LOCA are mitigated by the standby gas treatment system and held below 1 inch of water as discussed in Subsection 6.2.3. The pressure transient resulting from a HELB is documented in Reference 16. The peak pressures are shown in Reference 16.

b. Temperature

The temperature transient during a LOCA is not specifically evaluated as a part of the analysis discussed in Subsection 6.2.3. The temperature transient is chosen conservatively to be that of the LPCS pump cubicle as shown in Figures T34 (Curve 2) and T36 (Curve 4) of Reference 16.

The temperature following a HELB in the adjacent ECCS pump cubicles is shown in Reference 16.

c. Relative humidity

The relative humidity is estimated conservatively to be 100% for all accident conditions.

d. Duration

The LOCA pressure transient is inconsequential. The LOCA temperature transients are taken conservatively to be that of the LPCS pump cubicle as given in Figures 34 (Curve 2) and T36 (Curve 4) of Reference 16. The temperature at the end of Curve 4 shown in Figure T36 of Reference 16 is conservatively assumed to persist up to 100 days.

The conditions resulting from a HELB in adjacent ECCS pump cubicles are taken conservatively to be the values stated in Reference 16 for the duration of the transient.

3.11.9.20 Environmental Zone H-18

Zone H-18 is the RCIC pipe tunnel that is designated as area A.2.3 in Drawing M01-1600-7. The bounding environmental conditions for this zone result from a LOCA or a HELB in the main steam tunnel in the auxiliary building.

a. Pressure

The pressure transient for the LOCA is mitigated by the standby gas treatment system and held below 1 inch of water as discussed in Subsection 6.2.3.

The pressure transient for a HELB in the adjacent main steam tunnel is determined in Reference 11. The peak pressure determined by this analysis, 8.2 psig, is assumed to persist for the duration of the transient.

b. Temperature

The temperature transient for the LOCA conditions is calculated as discussed in Subsection 6.2.3.

The temperature for the HELB condition is shown in the bounding temperature curves provided in Reference 16. The temperature transient following a LOCA is controlled by the operation of the SGTS and is given in Figures T34 (Curve 3) and T35 (Curve 1) of Reference 16.

c. Relative humidity

For the LOCA condition the relative humidity is assumed conservatively to be 100%. For the HELB condition the environment is considered to be all steam.

d. Duration

The pressure transient for the LOCA condition is inconsequential. The temperature transient is given in Figures T34 (Curve 3) and T35 (Curve 1) of Reference 16. The temperature at the end of Curve 1 shown in Figure T35 is conservatively assumed to persist up to 100 days.

The temperature time-history for the HELB condition is analyzed or extrapolated for 100 days.

3.11.9.21 Environmental Zone H-19

This zone is the personnel hatch access area denoted as area A.2.5 in Drawing M01-1600-7. The bounding environmental conditions for this zone result from a HELB in the adjacent RWCU pump cubicles or a LOCA.

a. Pressure

The pressure transient resulting from a LOCA is mitigated by the standby gas treatment system and the peak pressure is held below 1 inch of water as discussed in Subsection 6.2.3.

The pressure transients resulting from a HELB in the adjacent RWCU pump cubicles are determined from Reference 9. The peak pressure value appears in Reference 16.

b. Temperature

The temperature transient during a LOCA is not specifically evaluated as part of the analysis discussed in Subsection 6.2.3. The temperature is conservatively chosen to be that of the adjacent LPCS pump cubicle as shown in Figures T34 (Curve 2) and T36 (Curve 4) of Reference 16.

The temperature following a HELB in the adjacent RWCU pump cubicles is provided in Reference 16.

c. Relative humidity

The relative humidity is estimated conservatively to be 100% for all accident conditions.

d. Duration

The LOCA pressure transient is inconsequential. The LOCA temperature transient is taken conservatively to be that of the LPCS pump cubicle as given in Figures T34 (Curve 2) and T36 (Curve 4) of Reference 16. The temperature at the end of Curve 4 shown in Figure T36 is conservatively assumed to persist up to 100 days.

The conditions resulting from a HELB in the adjacent RWCU pump cubicles are taken conservatively to be the values stated in Reference 16 for the duration of the transient.

3.11.9.22 Environmental Zone H-20

Zone H-20 is the auxiliary building pipe tunnel designated area A.2.6 in Drawing M01-1600-7. The bounding environmental conditions for this zone result from a RWCU line break in the tunnel or from a LOCA.

a. Pressure

The pressure transient following a LOCA is mitigated by the standby gas treatment system and held below 1 inch of water as discussed in Section 6.2.3.

The pressure transient following a HELB in the tunnel is taken from Reference 9. The peak pressure value is stated in Reference 16.

b. Temperature

The temperature transient following a LOCA is not specifically evaluated. Conservatively, the temperature transient in the LPCS pump cubicle is assumed. This transient is given in Figures T34 (Curve 2) and T36 (Curve 4) of Reference 16.

The temperature following a HELB is provided in Reference 16.

c. Relative humidity

The relative humidity is assumed conservatively to be 100% following a LOCA. The environment is considered as all steam following the HELB.

d. Duration

The pressure transient following a LOCA is inconsequential as discussed in Subsection 6.2.3. The temperature time-history is assumed conservatively to be that of the LPCS cubicle as shown in Figures T34 (Curve 2) and T36 (Curve 4) of Reference 16. The temperature at the end of Curve 4 shown in Figure T36 is conservatively assumed to persist up to 100 days.

The peak values for a HELB stated in Reference 16 are assumed to persist for the duration of the accident.

3.11.9.23 Environmental Zone H-21

Zone H-21 is composed of storage rooms and a stairway designated areas A.2.8, A.2.15, and A.2.21 in Drawing M01-1600-7. The bounding environmental conditions result from a LOOP.

a. Pressure

The pressure transient following a LOOP is provided in Table T10 of Reference 16.

b. Temperature

The temperature transients following a LOCA or LOOP are provided in Reference 16.

c. Relative humidity

The relative humidity following a LOOP is provided in Table T10 of Reference 16.

d. Duration

Time-histories for the LOOP temperatures are presented in Reference 16.

3.11.9.24 Environmental Zone H-22

Zone H-22 is composed of the air locks designated as areas A.2.9 and A.2.19 in Drawing M01-1600-7. The bounding environmental conditions result from a HELB in adjacent RHR or RWCU pump cubicles or from a LOCA.

a. Pressure

The pressure transient following a LOCA is mitigated by the standby gas treatment system as discussed in Subsection 6.2.3. The pressure transient following a HELB results in the peak pressures stated in Reference 16. Those pressures are based on the analysis given in Reference 9.

b. Temperature

The temperature transient following a LOCA is not specifically evaluated for these areas. The temperature transients are assumed conservatively to be that of the adjacent ECCS pump cubicles as determined in Subsection 6.2.3.

The temperature following a HELB is shown in Reference 16.

c. Relative humidity

The relative humidity is assumed conservatively to be 100% following any accident.

d. Duration

The pressure transient following a LOCA is inconsequential. The temperature transient is assumed conservatively to be that of the adjacent ECCS pump cubicles as given in Figures T34 (Curve 2) and T36 (Curve 4) of Reference 16. The temperature at the end of Curve 4 shown in Figure T36 is conservatively assumed to persist up to 100 days.

3.11.9.25 Environmental Zone H-23

Zone H-23 is the MSIV room access area designated as A.2.12 in Drawing M01-1600-7. The bounding environmental conditions in this zone result from a LOCA or a HELB in the adjacent Auxiliary Building steam tunnel.

a. Pressure

The pressure transient following a LOCA is mitigated by the standby gas treatment system. The peak pressure is held below 1 inch of water as stated in Subsection 6.2.3.

The peak pressure following a HELB in the adjacent RCIC pipe tunnel is stated in Reference 16. This pressure is determined from the analysis of Reference 9.

b. Temperature

The temperature transient following a LOCA is discussed in Subsection 6.2.3. Figures T34 (Curve 2) and T36 (Curve 4) of Reference 16 present the time-dependent temperature profiles.

The temperature following a HELB in the adjacent RCIC cubicle is shown in Reference 16.

c. Relative humidity

The relative humidity following any accident is assumed to be 100%.

d. Duration

The pressure transient following a LOCA is inconsequential. The temperature transient is given in Figures T34 (Curve 2) and T36 (Curve 4) of Reference 16. The temperature at the end of Curve 4 shown in Figure T36 is conservatively assumed to persist up to 100 days.

The accident conditions stated in Reference 16 are presumed to persist for the duration of the HELB accident.

3.11.9.26 Environmental Zone H-24

Zone H-24 is composed of the MSIV cubicles shown as areas A.2.13 and A.2.14 in Drawing M01-1600-7. The bounding environmental conditions for these cubicles result from a LOCA, LOOP, Loss of HVAC, or a HELB in the adjacent Auxiliary Building steam tunnel.

a. Pressure

Same as Zone H-23 during LOCA and HELB. The value for LOOP is the same as that for LOCA. The zone pressure is within its normal pressure range during a Loss-of-HVAC (see Reference 16).

b. Temperature

Same as Zone H-23 during LOCA and HELB. The value for LOOP is the same as that for LOCA. The peak temperatures for the cubicles in Zone H-24 are given in Reference 16.

c. Relative Humidity

Same as Zone H-23 during LOCA and HELB. The value for LOOP is the same as that for LOCA. The range of relative humidity for the cubicles in Zone H-24 are given in Reference 16.

d. Duration

Same as Zone H-23 during LOCA and HELB. The value for LOOP is the same as that for LOCA. Environmental conditions due to Loss-of-HVAC are assumed to persist for 7 days.

3.11.9.27 Environmental Zone H-25

Zone H-25 is composed of the RWCU pump cubicles shown as areas A.2.16, A.2.17, and A.2.18 in Drawing M01-1600-7. The bounding environmental conditions in this zone result from a LOCA or from a HELB within the cubicles.

a. Pressure

The pressure following a LOCA is mitigated by the standby gas treatment system. The pressure is held below 1 inch of water as discussed in Subsection 6.2.3.

The pressure following a HELB is documented in Reference 16. The peak pressures determined by this analysis are given in Reference 16.

b. Temperature

The temperature for the LOCA condition was not specifically evaluated for these cubicles. It is assumed conservatively that the temperature transient for the LPCS pump cubicle applies to these cubicles. The temperature following a HELB does not exceed 212°F except for a short-term spike of 214°F (< 60 seconds) and is provided in Reference 16.

c. Relative humidity

The relative humidity is conservatively assumed to be 100% following a LOCA. Following a HELB, the relative humidity is assumed to be 100%.

d. Duration

The pressure transient following a LOCA is inconsequential as discussed in Subsection 6.2.3. The temperature transients following a LOCA are conservatively assumed to be the same as those for the LPCS pump cubicle given in Figures T34 (Curve 2) and T36 (Curve 4) of Reference 16. The temperatures at the end of the curves shown in Figure T36 are conservatively assumed to persist up to 100 days.

The conditions stated in Reference 16 for the HELB are presumed to persist for the duration of the accident.

3.11.9.28 Environmental Zone H-26

Zone H-26 is the containment above the HCU floor elevation. These areas are designated C.3.1, C.4.1, and C.5.1 in Drawing M01-1600 Sheets 8, 9, and 10. The bounding environmental conditions result from design-basis LOCA events.

a. Pressure

Same as Zone H-15.

b. Temperature

Same as Zone H-15.

c. Relative humidity

Same as Zone H-15.

d. Duration

Same as Zone H-15.

e. Submergence or Spray

Same as Zone H-15 without pool swell.

3.11.9.29 Environmental Zone H-27

Zone H-27 is the drywell proper at the core midplane. Area C.3.2 in Drawing M01-1600-8 corresponds to this zone.

a. Pressure

Same as Zone H-2.

b. Temperature

Same as Zone H-2.

c. Relative humidity

Same as Zone H-2.

d. Duration

Same as Zone H-2.

3.11.9.30 Environmental Zone H-28

Zone H-28 is the annular area between the RPV and the biological shield wall noted as areas C.3.3 and C.4.3 in Drawing M01-1600 Sheets 8 and 9. The bounding environmental conditions in this zone result from a LOCA inside the annular area that defines the zone or a small line break HELB (LOCA) in the drywell per Zone H-2.

a. Pressure

The pressure transient for the HELB (LOCA) events in the drywell is the same as for Zone H-27 as given in Table T1 of Reference 16.

The pressure transients following a recirculation or feedwater line break inside the annulus are stated in terms of the peak pressures in Table T6 of Reference 16. The analysis by which these pressures were determined is discussed in Subsection 6.2.1.2.

b. Temperature

The temperature transient following the HELB (LOCA) events in the drywell is the same as for Zone H27 as given in Table T1 of Reference 16.

The temperature transients following the HELB (LOCA) events inside the annulus are conservatively assumed to be the saturation temperatures for the peak pressures determined in Subsection 6.2.1.2. These temperatures are tabulated in Table T6 of Reference 16.

c. Relative humidity

The relative humidity following a HELB (LOCA) in the drywell is given in Table T1 of Reference 16. The relative humidity following a HELB (LOCA) in the RPV annulus is given in Table T6 of Reference 16.

d. Duration

The time-histories for HELB (LOCA) events in the drywell are given in Table T1 of Reference 16. The peak conditions for the HELB (LOCA) events inside the annulus as given in Table T6 of Reference 16 are assumed to persist for the duration of the transients as discussed in Subsection 6.2.1.2.

3.11.9.31 Environmental Zone H-29

Zone H-29 is the main steam pipe tunnel in the containment. This is designated area C.3.4 in Drawing M01-1600-8. The bounding environmental conditions for this zone result from a LOCA or a RWCU line break in the tunnel. Main steam and feedwater line breaks are not considered because these lines are enclosed within guard pipes.

a. Pressure

The pressure transient following a LOCA is discussed in Section 6.2. The bounding environmental conditions following a LOCA are specified from the generic Mark III evaluation of Reference 7 and the envelopes of Reference 8.

The pressure transient following a HELB is discussed in Subsection 6.2.1.2.3.3.

b. Temperature

The temperature transient following a LOCA is discussed in Section 6.2. The bounding environmental conditions following a LOCA are specified from the generic analysis of Reference 7 and the envelopes of Reference 8.

The temperature transient following a HELB does not exceed 219°F and is shown in Reference 16.

c. Relative humidity

The atmosphere in the zone is conservatively assumed to have a relative humidity of 100% following a LOCA and to be all steam following a HELB. These values are given in Reference 16 and Table T4 of Reference 16.

The atmosphere in this zone is conservatively assumed to be all steam following a HELB (see Reference 16).

d. Duration

The time-histories for the LOCA events are given in Table T4 of Reference 16. The pressure and temperature time histories for the HELB are analyzed for 100 days or until conditions are restored to normal.

3.11.9.32 Environmental Zone H-30

Zone H-30 is the main steam tunnel in the auxiliary building. This zone is denoted as areas A.3.3 and A.4.3 in Drawing M01-1600 Sheets 8 and 9. The bounding environmental conditions in this zone result from a LOCA or a HELB in the tunnel.

a. Pressure

For LOCA, same as Zone H-18.

The pressure transient for a HELB in the main steam tunnel is determined in Reference 11. The peak pressure of 13.8 psig determined by this analysis is assumed to persist for the duration of the transient.

b. Temperature

Same as Zone H-18.

c. Relative humidity

Same as Zone H-18.

d. Duration

Same as Zone H-18.

3.11.9.33 Environmental Zone H-31

Zone H-31 is composed of the areas in the auxiliary building switchgear rooms that are subject to elevated radiation after a LOCA. The bounding environmental conditions in these areas, shown on Drawing M01-1600 Sheets 8 and 9 as A.3.6, A.3.7, A.4.6, and A.4.7, result from normal operating conditions for these areas. The radiation, which is the only parameter that changes after an accident, is expressed as total integrated dose.

a. Pressure

The areas in this zone are outside of the primary containment and thus receive no pressure effects due to a LOCA.

b. Temperature

The areas in this zone are outside of primary containment and thus receive no temperature effects due to a LOCA.

c Relative Humidity

The areas in this zone are outside of primary containment and thus experience no change in relative humidity due to a LOCA.

d. Duration

This is not applicable.

3.11.9.34 Environmental Zone H-32

Zone H-32 is the drywell proper above a floor elevation of 778 feet. This zone is designated C.4.2 in Drawing M01-1600-9.

a. Pressure

Same as Zone H-2.

b. Temperature

Same as Zone H-2.

c. Relative humidity

Same as Zone H-2.

d. Duration

Same as Zone H-2.

3.11.9.35 Environmental Zone H-33

Zone H-33 is the pipe tunnel designated as area C.4.4 in Drawing M01-1600-9. The bounding environmental conditions for this tunnel result from a LOCA and/or a HELB inside the tunnel.

a. Pressure

The LOCA pressure conditions are determined from the generic analysis of Reference 7 and the envelopes of Reference 8. The bounding pressure transients are given in Table T4 of Reference 16.

The pressure in area C.4.4 due to a HELB in the RWCU system remains nearly constant (0 psig) throughout the transient.

b. Temperature

The LOCA temperature conditions are determined from the generic analysis of Reference 7 and the envelopes of Reference 8. The bounding temperature transients are given in Table T4 of Reference 16.

The temperature following a HELB does not increase significantly from the normal temperature.

c. Relative humidity

The relative humidity following a LOCA/HELB is assumed conservatively to be 100%.

d. Duration

The time-histories for the LOCA events are provided in Table T4 of Reference 16. The temperature time-history for the HELB event is provided analyzed for 100 days.

3.11.9.36 Environmental Zone H-34

Zone H-34 is the RWCU backwash receiving tank cubicle shown as area C.4.5 in Drawing M01-1600-9.

a. Pressure

Same as Zone H-33, except refer to Table T5 instead of T4 of Reference 16.

b. Temperature

Same as Zone H-33, except refer to Table T5 instead of T4 of Reference 16.

c. Relative humidity

Same as Zone H-33, except refer to Table T5 instead of T4 of Reference 16.

d. Duration

Same as Zone H-33, except refer to Table T5 instead of T4 of Reference 16.

3.11.9.37 Environmental Zone H-35

Zone H-35 is composed of the filter/demineralizer recirculating pump cubicles and the pipe tunnel designated as areas C.4.7 and C.5.8 in Drawing M01-1600 Sheets 9 and 10. The bounding environmental conditions in this zone result from a LOCA or a HELB.

a. Pressure

The bounding environmental conditions in area C.4.7 are the same as those for area C.4.5 in Zone H-34. The response of area C.5.8 to a LOCA is the same as that of area C.4.5 in Zone H-34. The pressure in area C.5.8 due to a HELB in the RWCU valve room is assumed to remain at its peak value of 4 psig throughout the transient.

b. Temperature

The response of area C.4.7 to a LOCA or a HELB is the same as that of area C.4.5 in Zone H-34. The response of area C.5.8 to a LOCA is also the same as the response of area C.4.5 in Zone H-34. The temperature in area C.5.8 due to a HELB in the RWCU valve room is shown in Reference 16.

c. Relative humidity

The relative humidity following a LOCA is assumed conservatively to be 100% for area C.4.5 in Zone H-34. The environment is considered all steam following HELB for area C.5.8 in Zone H-34.

d. Duration

The time histories for the LOCA events are provided in Table T5 of Reference 16. The temperature for the HELB event are provided analyzed for 100 days.

e. Submergence or Spray

Flooding is discussed in Subsection D3.6.4.

3.11.9.38 Environmental Zone H-36

Zone H-36 is the RWCU crossover pipe tunnel shown as area C.5.11 in Drawing M01-1600-10. The bounding environmental conditions for this zone result from a spectrum of LOCA events in the drywell consisting of the large HELB or the small HELB as described in Table T5 of Reference 16 or a HELB within the pipe tunnel.

a. Pressure

The pressure transient following a LOCA is defined by the generic evaluation in Reference 7 and the envelopes of Reference 8. These values are shown in Table T5 of Reference 16.

The analysis for the pressure following a HELB is discussed in Section 6.2.1.2.3.6.

b. Temperature

The temperature transient following a LOCA is defined by the generic evaluation in Reference 7 and the envelopes of Reference 8. The values are given in Table T5 of Reference 16. The HELB conditions are determined as the saturation temperature for the pressure determined above.

c. Relative humidity

The relative humidity is provided in Table T5 of Reference 16.

d. Duration

The time history for the LOCA events are given in Table T5 of Reference 16.

3.11.9.39 Environmental Zone H-37

Zone H-37 is the containment building general area above an Elevation of 856 feet, 0 inches. Zone H-37, area C.7.1, is shown on Drawing M01-1600-11.

a. Pressure

Same as Zone H-15.

b. <u>Temperature</u>

Same as Zone H-15. The upper region of this zone is subject to a steam environment of approximately 220° F due to thermal stratification in the dome. This condition is conservatively assumed to persist for 100 days.

c. <u>Relative humidity</u>

Same as Zone H-15. The upper region of this zone is subject to a steam environment of approximately 220°F due to thermal stratification in the dome.

d. <u>Duration</u>

Same as Zone H-15. In addition, the upper region of this zone is subject to a steam environment of approximately 220° F due to thermal stratification in the dome. This condition is conservatively assumed to persist for 100 days.

e. <u>Submergence or Spray</u>

Equipment located in this zone is subject to the effects of the containment spray system operation.

3.11.9.40 <u>Environmental Zone H-38</u>

Zone H-38 is that portion of the gas control boundary that surrounds the containment structure above floor Elevation 801 feet, 9 inches shown as area C.5.2 in Drawing M01-1600 Sheets 10 and 11. The bounding environmental conditions in this zone result from a LOCA.

a. <u>Pressure</u>

The pressure environment for this zone is maintained below 1 inch of water by the standby gas treatments system following a LOCA. The analysis of this zone is discussed in Subsection 6.2.3.

b. <u>Temperature</u>

The temperature environment for this zone is not explicitly evaluated as part of the analysis discussed in Subsection 6.2.3. The temperature for this zone is conservatively chosen to be that of the fuel building following a LOCA. This environment is tabulated in Table T7 of Reference 16.

c. <u>Relative humidity</u>

The relative humidity is given in Table T7 of Reference 16.

d. <u>Duration</u>

The pressure transient following a LOCA is inconsequential. The temperature transient is conservatively assumed to be that of the fuel building as shown in Table T7 of Reference 16.

3.11.9.41 Environmental Zone H-39

Zone H-39 consists of the RWCU regenerative and nonregenerative heat exchanger cubicles shown as areas C.5.3 and C.5.4 in Drawing M01-1600 Sheets 9 and 10. The bounding environmental conditions for this zone result from a LOCA or a HELB in the cubicles.

a. Pressure

The pressure transient following a LOCA is determined from the generic analysis of Reference 7 and the envelopes of Reference 8. This transient is shown in Table T5 of Reference 16.

The HELB analysis is discussed in Subsection 6.2.1.2.

b. Temperature

The temperature transient following a LOCA is determined from the generic analysis of Reference 7 and the envelopes of Reference 8. This transient is given in Table T5 of Reference 16.

The temperature for the HELB is assumed to be the saturation temperature for the pressure described in Section 6.2.1.2.

c. Relative humidity

The relative humidity is assumed conservatively to be 100% following a LOCA. Following a HELB the environment is assumed to be all steam.

d. Duration

The accident transient conditions are analyzed or extrapolated for 100 days.

3.11.9.42 Environmental Zone H-40

Zone H-40 is the combustible gas control equipment cubicles shown as areas C.4.8 in Drawing M01-1600-9 and C.5.5 in Drawing M01-1600-10. The bounding environmental conditions for this zone result from a LOCA.

a. Pressure

Table T5 of Reference 16 gives the pressure values.

b. Temperature

Table T5 of Reference 16 gives the temperature values.

c. Relative humidity

Table T5 of Reference 16 gives the relative humidity values.

d. Duration

Table T5 of Reference 16 gives the duration.

3.11.9.43 Environmental Zone H-41

Zone H-41 is the filter/demineralizer holding pump cubicle shown as area C.5.6 in Drawing M01-1600-10. The bounding conditions in this zone result from a LOCA or a HELB within the cubicle.

a. Pressure

Same as Zone H-36, except refer to Table T4 instead of Table T5 of Reference 16, and refer to Section 6.2.1.2.3.7 instead of Section 6.2.1.2.3.6.

b. Temperature

Same as Zone H-36, except refer to Table T4 instead of Table T5 of Reference 16, and use the saturation temperature for pressure in Section 6.2.1.2.3.7 instead of Section 6.2.1.2.3.6.

c. Relative humidity

Same as Zone H-36.

d. Duration

Same as Zone H-36, except refer to Table T4 instead of Table T5 of Reference 16. Conditions following a HELB are analyzed or extrapolated for 100 days.

3.11.9.44 Environmental Zone H-42

Zone H-42 is the filter/demineralizer valve room shown as area C.5.7 in Drawing M01-1600-10. The bounding environmental conditions for this zone result from a LOCA or a HELB in the cubicles.

a. Pressure

Same as Zone H-36, except refer to Section 6.2.1.2.3.9 instead of Section 6.2.1.2.3.6.

b. Temperature

Same as Zone H-36, except use the saturation temperature for pressure described in Section 6.2.1.2.3.9.

c. Relative humidity

Relative humidity following LOCA and HELB is provided in Reference 16.

d. Duration

Same as Zone H-36.

3.11.9.45 Environmental Zone H-43

Zone H-43 is composed of the filter/demineralizer cubicles shown as areas C.5.9 and C.5.10 in Drawing M01-1600-10. The bounding environmental conditions for this zone result from a LOCA or a HELB within the cubicles.

a. Pressure

The pressure-time histories for areas C.5.9 and C.5.10 for the LOCA events are provided in Table T5 of Reference 16. The pressure-time histories for areas C.5.9 and C.5.10 due to the HELB events are discussed in Section 6.2.1.2.3.8.

b. Temperature

The response of areas C.5.9 and C.5.10 to the LOCA events is shown in Table T5 of Reference 16. The temperature response of these areas to the HELB event is shown in Reference 16.

c. Relative humidity

The relative humidity is provided in Reference 16.

d. Duration

The time histories for the LOCA events are provided in Table T5 of Reference 16. The time histories for the HELB events are provided in Reference 16. The conditions that exist at the end of these curves are assumed to persist for 100 days.

3.11.9.46 Environmental Zone H-44

Zone H-44 is the containment general area at the refueling floor. This zone is identified as area C.6.1 in Drawing M01-1600-11. The bounding environmental conditions for this zone result from design basis LOCA events.

a. Pressure

Same as Zone H-15.

b. Temperature

Same as Zone H-15.

c. Relative humidity

Same as Zone H-15.

d. Duration

Same as Zone H-15.

e. Same as Zone H-15 without pool swell.

3.11.9.47 Environmental Zone H-45

Zone H-45 is composed of the RWCU heat exchanger valve cubicles shown as areas C.5.12 and C.5.13 in Drawing M01-1600 Sheets 9 and 10. The bounding environmental conditions for these cubicles result from a LOCA or a HELB in the cubicles.

a. Pressure

Same as Zone H-36, except the analysis is discussed in Sections 6.2.1.2.3.5.

b. Temperature

Same as Zone H-36, except refer to Section 6.2.1.2.3.5 to determine the pressure to be used to determine saturation temperatures.

c. Relative humidity

Relative humidity is provided in Reference 16.

d. Duration

The time history for the LOCA events are given in Table T5 of Reference 16.

The pressure and temperature following a HELB event are analyzed or extrapolated for 100 days.

3.11.9.48 Environmental Zone H-46

Zone H-46 is composed of the hydrogen recombiner cubicles in the control building. These are shown as areas D.1.8 and D.1.9 in Drawing M01-1600-12. The bounding environmental conditions for this zone result from the operation of the hydrogen recombiners following a LOCA.

a. Pressure

Reference 16 gives the envelope for the pressure conditions following a LOCA.

b. Temperature

Reference 16 gives the envelope for the temperature conditions following a LOCA.

c. Relative humidity

Reference 16 gives the relative humidity following a LOCA.

d. Duration

The LOCA conditions stated in Reference 16 are applied for 100 days following the accident.

3.11.9.49 Environmental Zone H-47

Zone H-47 is composed of the standby gas treatment system filter train cubicles shown as areas D.2.8 and D.2.9 in Drawing M01-1600-13. The bounding environmental conditions in this zone result from the operation of the standby gas treatment system after a LOCA.

a. Pressure

Reference 16 gives the envelope for the pressure conditions following a LOCA.

b. Temperature

Reference 16 gives the envelope for the temperature conditions following a LOCA.

c. Relative humidity

Reference 16 gives the relative humidity following a LOCA.

d. Duration

The LOCA conditions stated in Reference 16 are applied for 100 days following the accident.

3.11.9.50 Environmental Zone H-48

This environmental zone is not used.

3.11.9.51 Environmental Zone H-49

This environmental zone is not used.

3.11.9.52 Environmental Zone H-50

This environmental zone is not used.

3.11.9.53 Environmental Zone H-51

Zone H-51 is the control building general area shown as D.2.10 in Drawing M01-1600-13. This area experiences an elevated radiation level following LOCA but the LOCA produces no other change in the environmental parameters. Bounding environmental conditions are produced by a LOOP transient in the area.

a. Pressure

There is no significant pressure transient in the area following a LOOP.

b. Temperature

Summer and winter transients were calculated to provide a temperature envelope for these areas. The extreme temperature for the summer condition was reached during the 100-day transient following a LOOP. For the winter transient there was a slight increase in the temperature at the end of 100 days. The initial condition in the area represents the minimum temperature condition. Temperature extremes are presented in Reference 16.

c. Relative Humidity

The high normal relative humidity and the low normal relative humidity were used, respectively, as initial conditions for the winter and summer transients to provide a bounding relative humidity envelope. Humidity extremes are presented in Reference 16.

d. Duration

Time histories for the summer and winter LOOP temperatures are analyzed for 100 days in Figures T14 and T15 of Reference 16.

3.11.9.54 Environmental Zone H-52

Zone H-52 is the radwaste pipe tunnel in the auxiliary building. This zone is identified as area A.2.2 in Drawing M01-1600-7. The bounding environmental conditions result from a LOCA or HELB within the tunnel.

a. Pressure

The pressure transient following a LOCA is mitigated by the standby gas treatment system and held below 1 inch of water. See Subsection 6.2.3 for the supporting analysis.

The pressure following a HELB within the sub-compartment results in the peak pressure stated in Reference 16. This pressure is based on the analysis in Reference 9.

b. Temperature

The determination of the temperature for this area following a LOCA has not been specifically determined as part of the analysis presented in Subsection 6.2.3. However, the use of the temperature curves for the LPCS pump room profiles that appear in Figures T34 (Curve 2) and T36 (Curve 4) of Reference 16 are conservative for this zone.

The temperature following a HELB is assumed conservatively to be the saturation temperature for the peak pressure given in Reference 16.

c. Relative humidity

Per Figure T21 of Reference 16 the relative humidity following LOCA does not rise significantly above the normal maximum but assumed to be 100%. The environment following a HELB is assumed to be all steam.

d. Duration

The pressure transient following a LOCA is inconsequential. The temperature transient is given in Figures T34 (Curve 2) and T36 (Curve 4) of Reference 16. The conditions given at the end of Curve 4 in Figure T36 are conservatively applied over the period from 30 to 100 days.

3.11.9.55 Environmental Zone H-53

Zone H-53 is the drywell. The zone is denoted as area C.2.3 in Drawing M01-1600-7. The bounding environmental conditions for this zone result from design-basis LOCA events. For abnormal conditions during scram refer to Figure T37 of Reference 16.

a. Pressure

Same as Zone H-16.

b. Temperature

Same as Zone H-16.

c. Relative humidity

Same as Zone H-16.

d. Duration

Same as Zone H-16.

e. Submergence or Spray

Same as Zone H-16.

3.11.9.56 Environmental Zone H-54

Zone H-54 is the cubicle in the turbine building containing the turbine lube oil reservoir. This zone is denoted as area T.3.1 in Drawing M01-1600-21. The bounding environmental conditions in this zone results from a HELB involving a high energy instrument line.

a. Pressure

This zone will experience a negligible pressure transient due to a HELB in a main steam instrument line. The peak pressure is given in Reference 16.

b. Temperature

The peak temperature is conservatively chosen as that resulting from an isenthalpic expansion from reactor pressure to the cubicle pressure. The peak temperature is specified in Reference 16.

c. Relative humidity

The relative humidity for this zone is given in Reference 16.

d. Duration

The cubicle would stay at peak temperature and humidity conditions for 100 days or until isolation of the break is achieved.

3.11.9.57 Environmental Zone H-55

Zone H-55 is the head cavity which is above, and separated from the drywell by the refueling bulkhead. The other boundaries are formed by the drywell head and the RPV head. There are ventilation paths between the head cavity and the rest of the drywell. This zone is denoted as area C.5.14 in Drawing M01-1600-10. The bounding environmental temperature and humidity conditions result from a small HELB inside the head cavity. A large HELB outside the head cavity provides the bounding pressure condition during the initial portion of the transient.

a. Pressure

This zone pressurizes to 30 psig in 1.5 seconds following a large HELB outside the head cavity and remains at this level during the first 40 seconds of the transient. During the next 5 seconds the pressure drops to 15 psig. After the first 45 seconds of the transient the bounding pressure environment is that for a small HELB inside the head cavity as given in Table T11 of Reference 16.

b. Temperature

The bounding environmental temperature condition is that for a small HELB inside the head cavity. Within 0.5 second after the break a peak temperature of 339.9°F is reached. The temperature transient is given in Table T11 of Reference 16.

c. Relative Humidity

The bounding humidity environmental condition is that for a small HELB inside the head cavity in which an all steam environment is specified for the first 6 hours of the transient followed by 100% relative humidity for the rest of the 100 day duration.

d. Duration

Durations are as specified in Table T11 of Reference 16 for the bounding line breaks as described above.

e. Submergence or Spray

The refueling bulkhead is designed to allow flooding of the head area during refueling. All vents have watertight caps which are closed to allow flooding of the area.

3.11.9.58 Environmental Zone H-56

Zone H-56 denotes one general area in the auxiliary building outside secondary containment. The area borders on RHR pump rooms A and B and is designated A.2.7 in Drawing M01-1600-7. The bounding environmental conditions are produced by a LOOP transient in this zone.

a. Pressure

The pressure is not likely to deviate significantly from the normal range as shown in Table T10 of Reference 16.

b. Temperature

Summer and winter transients were calculated to provide a temperature envelope for the area. Temperature extremes were reached during the 100 day duration considered following a LOOP. These temperatures are presented in Reference 16.

c. Relative Humidity

The high normal relative humidity and the low normal relative humidity were used respectively, as initial conditions for the winter and summer transients, to provide a bounding relative humidity envelope. Humidity extremes are presented in Reference 16.

d. Duration

The transient is assumed to last for the duration of the LOOP.

3.11.9.59 Environmental Zone H-57

Zone H-57 is the "control building general area" shown as D.2.11 and D.2.12 in Drawing M01-1600-13. This area experiences an elevated radiation level following LOCA but the LOCA produces no other change in the environmental parameters. Bounding environmental conditions are produced by a LOOP transient in the area.

a. Pressure

Same as Zone H-51.

b. Temperature

Same as Zone H-51.

c. Relative Humidity

Same as Zone H-51.

d. Duration

Same as Zone H-51.

3.11.9.60 Environmental Zone H-58

Zone H-58 denotes one general area in the auxiliary building outside secondary containment. The area is a corridor adjacent to the RCIC instrument panel room and is designated are A.1.12 in Drawing M01-1600-6. The bounding environmental conditions are produced by a LOOP transient in this zone.

a. Pressure

The pressure is not likely to deviate significantly from the normal range as shown in Table T10 of Reference 16.

b. Temperature

Summer and winter transients were calculated to provide a temperature envelope for this area. Temperature extremes were reached during the 100 day duration considered following a LOOP. These temperatures are presented in Reference 16.

c. Relative Humidity

The high normal relative humidity and the low normal relative humidity were used respectively, as initial conditions for the winter and summer transients, to provide a bounding relative humidity envelope. Humidity extremes are presented in Reference 16.

d. Duration

The transient is assumed to last for the duration of the LOOP.

3.11.9.61 Enviromental Zone H-59

Revision Zone H-59 denotes one general area in the auxiliary building outside secondary containment. The area borders on RHR pump room A and is designated A.1.15 in Drawing M01-1600-6. The bounding environmental conditions are produced by a LOOP transient in this zone.

a. Pressure

The pressure is not likely to deviate significantly from the normal range as shown in Table T10 of Reference 16.

b Temperature

Summer and winter transients were calculated to provide a temperature envelope for this area. Temperature extremes were reached during the 100 day duration considered following a LOOP. These temperatures are presented in Reference 16.

c. Relative Humidity

The high normal relative humidity and the low normal relative humidity were used respectively, as initial conditions for the winter and summer transients, to provide a bounding relative humidity envelope. Humidity extremes are presented in Reference 16.

d. Duration

The transient is assumed to last for the duration of the LOOP.

3.11.9.62 Enviromental Zone H-60

Zone H-60 is composed of one general area of the auxiliary building. This area is identifieed as A.1.16 in Drawing M01-1600-6. The bounding environmental conditions for this area result from a LOOP.

a. Pressure

The area in this zone is outside the secondary containment and thus is not subject to any pressure effects due to a LOCA. The pressure variations which would occur in this area during a LOOP transient are given in Table T10 of Reference 16.

b. Temperature

The area in this zone is outside the secondary containment and thus is not subject to the temperature effects due to a LOCA. The bounding temperature is determined as the result of a postulated LOOP. The extreme expected temperatures are cited in Table T10 of Reference 16.

c. Relative Humidity

The extreme values of relative humidity for this area are given in Table T10 of Reference 16.

d. Duration

The only transients of significance relate to the temperature. These are assumed to last for the duration of the LOOP.

3.11.10 Mild Environmental Zones

3.11.10.1 Temperature, Pressure, and Relative Humidity

The normal service conditions of the plant represent those conditions expected to occur during Planned Operation, with the normal HVAC systems in operation. The upper end of the temperature range represents the maximum value expected with all heat-generating equipment in the respective zones operating during a design-basis summer day. The lower end of the temperature range represents the minimum value expected during a design-basis winter day with equipment (except lighting) not operating and normal HVAC systems operating.

3.11.10.2 Mild Environment Radiation

The radiation environment for normal reactor operation is determined on the basis of the reactor operating at full power, fuel leaking at the design-basis level, and assuming that all the systems at the station are operating normally at full capacity. The normal radiation environment is reported as the integrated radiation dose that can be absorbed by components over the 40-year lifetime of the station. The upper limit of the total integrated dose is 10^4 Rad (C) for mild environments, except for solid state electronic devices for which the limit is 10^3 Rad (C).

3.11.11 Program for Continuing Qualification

As a part of the engineering review of each equipment qualification report, the qualified life of each equipment or component is identified, along with any maintenance requirements for maintaining this qualified life. This information is located in MS-02.00 (Reference 13) under List 2. The MS-02.00 Maintenance Standard consists of List 1 Equipment List, and List 2 Maintenance Requirement, and is maintained and controlled separately from the binders in PassPort as NSED – MEQPM-MS-02-00.

These replacement and maintenance requirements or changes to these requirements are processed through engineering change and will be integrated into the Clinton Power Station preventive maintenance program described in existing plant procedures. This preventive maintenance program includes the following features:

a. A computerized data base that includes performance frequency, assigned responsibility and applicable procedures, instructions, and requirements.

b. Periodic computer-generated schedules are produced and distributed to assigned personnel to perform maintenance checks.

c. Scheduled activities are performed and documented on schedules, data sheets, or other maintenance forms prescribed in procedures.

d. Completed schedules are used to update and provide feedback to the data base for rescheduling activities.

e. All documentation is retained for trend analysis and maintenance planning.

Maintenance actions resulting from Maintenance and/or Surveillance Testing are noted and result in the generation of a work document (either directly or subsequent to the generation of a

CPS Condition Report). Each work document generated for safety-related equipment is analyzed to determine if the required maintenance action was due to age-related degradation as a result of environmental conditions such as temperature, pressure, radiation, humidity, etc.

If a maintenance action is the result of age-related degradation, the Preventive Maintenance and/or Surveillance Testing schedules are reevaluated and modified, as necessary, to prevent future age-related degradation problems.

3.11.12 Central File Description

The Central File maintained at the Clinton plant site shall consist of the following items for Environmental Qualification of Electrical/Mechanical Equipment:

a. A copy of USAR Section 3.11 including all tables and figures.

b. System Description.

c. Environmental qualification (EQ) binders* for the equipment contained in List 1 of MS-02.00. Typically this would contain:

    1. Cover Page and Index Page

    2. Issue Summary Sheet

    3. Checklist for NUREG-0588 requirements (Tab A & Tab B).

    4. Maintenance and Surveillance Requirements (filed separately under List 2 of MS-02.00), Maintenance Standard.

    5. Qualification Test Report (Tab F).

    6. Analysis, Calculation, or Justification to support qualification (Tab C).

    7. Equipment Identification (Tab D).

    8. All supporting documents including excerpts from appropriate references (if required) (Tab G).

    9. System Components Evaluation Worksheets (SCEW) (Tab H).

d. Purchase order records pertaining to the Safety-Related equipment.

e. All other references pertaining to the equipment such as drawings, etc. that are not listed in items a through d above.

f. DC-ME-09-CP, Equipment Environmental Design Conditions Design Criteria Document.

* EQ binders for mechanical equipment are identified with prefix M to the binder numbers.

3.11.13 TMI Items Requiring Environmental Qualification

TMI items requiring environmental qualification such as area radiation monitors, miscellaneous recorders and transducers, etc., were added to List 1 of MS-02.00 as the systems were designed and the equipment purchased. Requirements resulting from the TMI-2 accident are addressed in Appendix D.

3.11.14 Mechanical Equipment Qualification

3.11.14.1 Introduction

Mechanical equipment which is environmentally qualified includes all safety-related active equipment located in harsh environmental zones. Other safety-related active mechanical equipment located in mild environmental zones and passive mechanical equipment is considered qualified by its construction as required by national codes, standards and NRC regulations.

Mechanical equipment which is environmentally qualified is listed in List 1 of MS-02.00 similar to Electrical Equipment in Section 3.11.8.

Components of the mechanical equipment listed in List 1 of MS-02.00 contain predominately metallic and some non-metallic materials. Since the effects of temperature, humidity and radiation are relatively insignificant for metallic components the environmental qualification is based only on their non-metallic materials.

Examples of components upon which qualification is based are:

- Gaskets
- Grease
- O-Rings
- Packing
- Diaphragms
- Seals
- Oil
- Hydraulic Fluids
- Lubricants

The evaluation of equipment for qualification is done with the aid of a checklist which address the following items:

1. Equipment name, number, vendor/manufacturer and model number.
2. Vendor's equipment qualification report number, reference and status.
3. Equipment description, including location.
4. References to standards and documents, where applicable, if requirements are met by the qualification.
5. Identifies method of qualification; (i.e. Operating Experience, Analysis, Certificate of Compliance, Type Testing, or a combination thereof).

3.11.14.2 Qualification Procedure

For mechanical equipment listed in List 1 of MS-02.00 a materials list of organic components is generated from drawings, catalogues or the equipment vendor's instruction manuals. Subsequently, an evaluation of the environmental impact on these organic components is made as follows:

a. Certificate of Compliance

Research procurement specifications to determine if environmental parameters for qualification of organic material were included in the equipment vendor's scope of work. If these parameters were specified, qualification is achieved through acquisition of a certificate of compliance from the equipment vendor. In some cases, the vendor's qualification report may also be reviewed to verify qualification.

b. Analysis

When environmental parameters are not contained in the procurement specification, qualification will be established by analysis of those organic materials whose degradation could affect the ability of the equipment to perform its safety related function. Such analysis will demonstrate similarity between qualified and unqualified materials and their function, or show material suitability through degradation analysis such as Arrhenius calculations or other suitable methods. The resulting analysis will be made part of the qualification documentation. The feasibility of this approach depends on the type of components (organics) and their composition.

c. Operating Experience

Qualification of mechanical equipment using operating experience is used as a basis for environmental qualification should a) and b) not be feasible. 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 done to qualify the equipment for the Design Basis Events. This information is documented in the environmental qualification documentation.

d. Type Testing

If environmental qualification of the item is not in the vendor's scope of work and b) and c) are not viable, the equipment is qualified by limited type testing of the components (organic) in question. A test specification is prepared either uniquely for Clinton or in a shared program with other utilities. The specification includes specific environmental qualification requirements and profiles for the equipment being qualified. Qualification results are reviewed and comments resolved prior to completing the equipment qualification package.

3.11.14.3 Central File Description

The qualification document is maintained in the Central File (filed by equipment qualification package) at the Clinton Plant Site and consist of the following items:

a. Individual component qualification documentation, (i.e. checklist and attachments) which identify conclusions of evaluation, equipment life, and maintenance requirements.

b. Verification and quality assurance records associated with equipment qualification.

3.11.15 References

1. "Commission Memorandum and Order," CL1-80-21, May 23, 1980.

2. "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," NUREG-0588 Revision 1, July, 1981.

3. "IEEE Standard for Qualifying IE Equipment for Nuclear Power Generating Stations," IEEE-323.

4. "IEEE Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations," IEEE-344.

5. "IEEE Standard for Design Qualification of Safety Systems Equipment Used in Nuclear Power Generating Stations," IEEE-627.

6. U.S. NRC letter to operating license applicants, "Qualification of Safety-Related Electrical Equipment," February 21, 1980.

7. "Containment Loads Appendix 3B to the 238 Nuclear Island GE Standard Safety Analysis Report - Submitted Amendment 2 to Application FDA review - STN 50-447," GE Document 22A7000, Revision 2.

8. "BWR Equipment Environmental Interface Data," GE Document 22A6926AA.

9. "High Energy Line Breaks Outside Containment," Sargent & Lundy Nuclear Safeguards and Licensing Division, Calculation 3C10-0477-001.

10. "Assessment of Weir Wall Annulus Flow During Drywell Depressurization," Sargent & Lundy Nuclear Safeguards and Licensing Division, Calculation 3C10-1280-001.

11. "Pressure and Temperature Effects of Postulated High Energy Line Breaks Outside of Containment," Sargent & Lundy Nuclear Safeguards and Licensing Division, Calculation 3C10-1274-001.

12. 10 CFR 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.

13. Nuclear Station Engineering Standard MS-02.00, Maintenance of Equipment Qualification Program Manual.

14. “Secondary Containment Functional Design,” Sargent & Lundy Calculation 3C10-1079-001.

15. “Secondary Containment Subcompartment Parameters For Environmental Qualification of Equipment,” Sargent & Lundy Calculation 3C10-0182-004.

16. DC-ME-09-CP, “Equipment Environmental Design Conditions,” Design Criteria Document.

17. Sargent & Lundy Letter #SLMI-8490 dated September 17, 1982 on Equipment Qualification Testing Containment Spray.

The following tables have been removed from the USAR.

Table 3.11-1	Environmental Qualification List - BOP Electrical Equipment
Table 3.11-2	Environmental Qualification List - NSSS Electrical Equipment
Table 3.11-3	Qualification Table for BOP Components
Table 3.11-4	Qualification Table for NSSS Components
Table 3.11-5	Environmental Zone Summary Table*
Table 3.11-6	Abnormal Conditions in Drywell*
Table 3.11-7	Abnormal Conditions in ECCS Pump Rooms*
Table 3.11-8	Environmental Conditions for ECCS Subareas in Secondary Gas Control Boundary*
Table 3.11-9	Deleted during FSAR Development
Table 3.11-10	Deleted during FSAR Development
Table 3.11-11	Abnormal Conditions in Containment*
Table 3.11-12	Abnormal Conditions in Cnmtainment Equipment Cubicles*
Table 3.11-13	Abnormal Conditions Between Shield Wall & RPV*
Table 3.11-14	Abnormal Conditions in Fuel Building*
Table 3.11-15	Deleted during FSAR Development
Table 3.11-16	Deleted during FSAR Development
Table 3.11-17	Deleted during FSAR Development
Table 3.11-18	Deleted during FSAR Development
Table 3.11-19	Deleted during FSAR Development
Table 3.11-20	Environmental Qualification List of Active NSSS & BOP Mechanical Equipment
Table 3.11-21	Time-Temperature Profile for Area F.3.1*
Table 3.11-22	Abnormal Conditions in CGCS Cubicles
Table 3.11-23	Environmental Zone Max & Min Temperature, Pressure, & Relative Humidity for Transient Conditions*
Table 3.11-24	Abnormal Conditions for Area Above RPV Head and Below Drywell Head*
Table 3.11-25	Abnormal Conditions in Drywell Under RPV*

* Information related to environmental conditions may be located in DC-ME-09-CP, Equipment Environmental Design Conditions Design Criteria Document.