SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 3
DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

SECTIONS

APPENDICES

3E	Procedure For Calculating Elasto-Plastically Designed Pipe Whip Restraint Loads By Equivalent Static Analysis Method
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3.1 CONFORMANCE TO NRC GENERAL DESIGN CRITERIA

This section discusses the extent to which the design criteria for plant structures, systems and components important to safety meet the NRC General Design Criteria for Nuclear Power Plants, specified in Appendix A to 10 CFR Part 50. For each criterion, a summary is provided which discusses how the principal design features meet the criterion and identifies any exceptions that are taken. In the discussion of each criterion, the chapters or sections of this Updated FSAR where more detailed information is presented are referenced to demonstrate compliance with the criterion.

3.1.1 Overall Requirements

3.1.1.1 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 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.

RESPONSE

The structures, systems and components of the facility are classified according to their importance in the prevention and mitigation of accidents, using the classification system developed by the American Nuclear Society. Each safety-related component is given a safety class designation. The codes, standards and quality controls applicable to each type of safety-related component and its safety class designation are identified in Chapter 3. Where applicable, design, fabrication, erection and testing are in accordance with the codes required in 10 CFR 50.55a.

Recognized codes and standards, when used, are identified and evaluated to assure their applicability, adequacy and sufficiency in keeping with the required safety function.

The quality assurance programs established by Yankee Atomic Electric Company, United Engineers and Constructors Inc., and Westinghouse conform with the requirements of 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, and are discussed in Chapter 17.

Chapter 14 describes initial tests and operation to assure performance on installed equipment commensurate with the importance of the safety function.

Component safety classifications are shown on P&IDs presented with their appropriate sections.

In accordance with the quality assurance programs, complete sets of records of the design, fabrication, construction and testing of safety-related components are maintained by the nuclear power licensee throughout the life of the plant.

3.1.1.2 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: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

RESPONSE

The structures, systems and components important to safety are designed to either withstand the effects of natural phenomena without loss of capability to perform their safety functions, or to fail in the safest condition. Those structures, systems and components vital to the safe shutdown capability of the reactor are designed to withstand the maximum probable natural phenomenon expected at the site, determined from recorded data for the site vicinity, with appropriate margin to account for uncertainties in historical data. Appropriate combinations of normal, accident and natural phenomena structural loadings are considered in the design of the structures, systems and components important to safety.

The nature and magnitudes of the natural phenomena considered in the design of this plant are discussed in Sections 2.3, 2.4, and 2.5. Sections 3.2 through 3.11 discuss the design of the plant in relationship to natural events. Seismic and safety classifications, as well as other pertinent standards and information, are given in the sections discussing individual structures and components.

3.1.1.3 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 fighting systems of appropriate capacity and capability shall be provided and designed to minimize the 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.

RESPONSE

The plant is designed to minimize the probability and effect of fires and explosions. Noncombustible and fire-resistant materials are used in the containment and control room, as well as in components of safety features systems and wherever practical throughout the plant.

Equipment and facilities for fire protection, including detection, alarm, and extinguishment are provided to protect both plant equipment and personnel from fire and explosion.

Fire-suppression systems are designed to assure that their rupture or inadvertent operation will not significantly impair systems important to safety.

The Fire Protection System is described in Chapter 9.

3.1.1.4 Criterion 4 - Environmental and Missile Design Bases

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including 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. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

RESPONSE

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. Criteria are presented in Chapter 3; environmental conditions are presented in Chapter 3 and 6.

These structures, systems, and components are protected, as appropriate, 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.

Details of the design, environmental testing, and construction of these systems, structures, and components are included in Chapters 3, 5, 6, 7 and 9.

Evaluation of the performance of safety features is contained in Chapter 15.

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

Structures, systems, and components important to safety shall not be shared among nuclear power plants 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.

RESPONSE

Since Seabrook Station is a single unit plant, there is no sharing of structures, systems and/or components as described in GDC-5.

Further discussion is presented in Chapters 6, 8, and 9.

3.1.2 Protection by Multiple Fission Product Barriers

3.1.2.1 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.

RESPONSE

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to:

Assure that fuel damage* is not expected during normal core operation and operational transients (Condition I)** or any transient conditions arising from occurrences of moderate frequency (Condition II)**. It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the Plant Cleanup System, and are consistent with plant design bases.

Ensure return of the reactor to a safe state following a Condition III** event with only a small fraction of fuel rods damaged, although sufficient fuel damage might occur to preclude resumption of operation.

Assure that the core is intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV) **.

Chapter 4 discusses the design bases and design evaluation of core components. Details of the control and protection systems instrumentation design and logic are discussed in Chapter 7. This information supports the accident analyses of Chapter 15, which show that the acceptable fuel design limits are not exceeded for Condition I and II occurrences.

3.1.2.2 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.

RESPONSE

Prompt compensatory reactivity feedback effects are assured when the reactor is critical by the negative fuel temperature effect (Doppler effect) and by the non-positive operational limit on moderator temperature coefficient of reactivity. A non-negative moderator temperature coefficient is allowed by Technical Specifications for all power levels, provided that compliance with the ATWS Rule and its basis are maintained, as described in the Bases for Technical Specification 3/4.1.1.3. The Seabrook core design philosophy meets this requirement by ensuring that a non-positive MTC exists for operating conditions above 20% power. The negative Doppler coefficient of reactivity is assured by the inherent design using low-enrichment fuel; the non-positive moderator temperature coefficient of reactivity is assured by administratively controlling the dissolved absorber concentration or by burnable poison.

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* Fuel damage as used here is defined as a penetration of the fission product barrier (i.e., the fuel rod clad).

** Defined by ANSI-N18.2-73.

These reactivity coefficients are discussed in Section 4.3.

3.1.2.3 Criterion 12 - Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control, and protection system 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.

RESPONSE

Power oscillations of the fundamental mode are inherently eliminated by the negative Doppler and non-positive moderator temperature coefficient of reactivity.

Oscillations, due to xenon spatial effects, in the radial, diametral, and azimuthal overtone modes are heavily damped due to the inherent design and due to the negative Doppler and non-positive moderator temperature coefficients of reactivity.

Oscillations, due to xenon spatial effects, in the axial first overtone mode may occur. Assurance that fuel design limits are not exceeded by xenon axial oscillations is provided by reactor trip functions using the measured axial power imbalance as an input.

Oscillations, due to xenon spatial effects, in axial modes higher than the first overtone, are heavily damped due to the inherent design and due to the negative Doppler coefficient of reactivity.

Xenon stability control is described in Section 4.3.

3.1.2.4 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 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.

RESPONSE

Instrumentation and controls are provided to monitor and control neutron flux, control rod position, temperatures, pressures, flows, and levels as necessary to assure that adequate plant safety can be maintained. Instrumentation is provided in the Reactor Coolant System, Steam and Power Conversion System, the Containment, Engineered Safety Features Systems, Radiological Waste Systems and other auxiliaries. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in the control room in proximity with the controls for maintaining the indicated parameter in the proper range.

Appropriate controls maintain these variables and systems within prescribed operating ranges.

The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems over the full design range of the plant. These systems are described in Chapter 6, 7, 8, 9, 11 and 12.

3.1.2.5 Criterion 14 - Reactor Coolant Pressure Boundary

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

RESPONSE

The Reactor Coolant System boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation, including all anticipated transients, and to maintain the stresses within applicable stress limits. See Section 3.9 for details. Reactor coolant pressure boundary materials, selection and fabrication techniques ensure a low probability of gross rupture or significant leakage.

In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as seismic and pipe rupture as discussed in Sections 3.6 and 3.7. The system is protected from overpressure by means of pressure relieving devices as required by applicable codes (see Subsection 5.2.2).

The Reactor Coolant System boundary has provisions for inspection, testing and surveillance of critical areas to assess the structural and leak-tight integrity. See Section 5.2 for details. For the reactor vessel, a material surveillance program conforming to applicable codes is provided. See Section 5.3 for details.

3.1.2.6 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 of normal operation, including anticipated operational occurrences.

RESPONSE

The design pressure and temperature for each component in the reactor coolant and associated auxiliary control, and protection systems are selected to be above the maximum coolant pressure and temperature under all normal and anticipated transient load conditions.

Additionally, reactor coolant pressure boundary components achieve a large margin of safety by using proven ASME materials and design codes, proven fabrication techniques, nondestructive shop testing, and integrated hydrostatic testing of assembled components. Chapter 5 discusses the reactor coolant system design.

3.1.2.7 Criterion 16 - Containment Design

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

RESPONSE

A reinforced concrete outer containment enclosure encloses the steel-lined reinforced concrete primary containment structure which, in turn, encloses the entire Reactor Coolant System. The primary containment structure is designed to sustain, without loss of integrity, all effects of the most limiting faults up to and including the double-ended rupture of the largest pipe in the Reactor Coolant System. The Emergency Core Cooling System and the Containment Spray System serve to cool the reactor core and the containment and return the containment to near atmospheric pressure following a LOCA.

Leakage from the primary containment is limited by design during the post LOCA pressurized period. Continued primary containment integrity is assured by periodic leak testing.

The secondary containment provides the ability to process any leakage from the primary containment, which results from the post LOCA differential pressure.

The containment and safety-related systems are designed to assure the required functional capability of containing any uncontrolled release of radioactivity, and no design conditions important to safety are exceeded for as long as postulated accident conditions require.

Refer to Chapters 1, 3, 6, 15 and 16.

3.1.2.8 Criterion 17 - Electrical Power Systems

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

The onsite electric power supplies, including the batteries, and the Onsite Electric Distribution System, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

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

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

RESPONSE

An onsite electric power system is provided and is designed with adequate independence, capacity, redundancy, and testability to assure the functioning of safety-related systems. Independence is provided by physical separation of components and cables to minimize vulnerability of redundant engineered safety features systems to single credible accidents.

Offsite power from the transmission network to the switching substation is supplied by three physically independent transmission lines. Power from the switching substation is provided for the safety features system of each unit through both the unit auxiliary transformers and reserve auxiliary transformers. These two transformer banks are independently connected to the switching substation and to each safety features load group to provide two circuits to power the two redundant load groups within a few seconds following a loss-of-coolant accident. A failure of a single component will not prevent safety-related systems from performing their function. Chapter 8 provides further details.

The onsite AC source of electrical power, described in Section 8.3, consists of two diesel generators, one connected to each of the redundant emergency buses of the unit. One diesel generator is capable of supplying sufficient power for the operation of the minimum safety features required for the unit during a postulated loss-of-coolant accident. However, during a postulated loss-of-coolant accident, each diesel generator starts automatically and, if offsite power is not available, it connects to its associated emergency bus. The safety features equipment is then sequentially started. The emergency buses and their associated diesel generators are so arranged that a failure of a single component will not prevent the power supply systems from performing their function. Chapter 8 provides further details.

Four DC batteries are provided in physically separated rooms, and two of these (one in each train) are adequate to supply the DC control power required for the safety features. For each of the four protection channels, an independent 120V AC vital power source is provided. Two additional vital 120V AC power sources are provided for other safety-related loads. Failure of a single component in this system will not impair control of the minimum safety features required to maintain the unit in a safe condition. Chapter 8 provides further details.

3.1.2.9 Criterion 18 - Inspection and Testing of Electrical Power Systems

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

RESPONSE

Class 1E systems are designed to permit appropriate periodic inspection and testing of:

a. Important areas and features such as wiring, insulation, connections, and switchboards during equipment shutdown.

b. The operability and functional performance of power system components, such as diesel generators, relays, buses, their DC system and circuit breakers and control circuits.

c. The operability of the electric power systems as a whole during plant shutdown.

d. The full operational sequence that brings the power systems into operation under conditions as close to design as practical, including operation of the Protection System and the transfer of power among the nuclear power unit, the preferred (offsite) power supply, and the standby (onsite) power supply systems.

Complete provisions for testing of Class 1E electric power systems and the standby power supplies (diesel generators) are described in Chapter 8.

3.1.2.10 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 protections 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 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.

RESPONSE

Safe occupancy capability of the control room during normal and accident conditions is provided in the design. The control room is in a seismic Category I structure. Adequate shielding is provided to maintain radiation levels in the control room below 5 rem whole body, or its equivalent to any part of the body, for the duration of a postulated design basis accident.

The control room is equipped with the primary and secondary control panels that contain those instruments and controls necessary to operate the plant safely under normal conditions and maintain it in a safe condition under accidents, including loss-of-coolant accidents.

In the unlikely event that access to the control room is restricted, local control stations and manual operation of critical components can be used to effect hot shutdown from outside the control room for an extended period.

By use of appropriate emergency procedures, the unit can also be brought to cold shutdown from outside the control room.

Refer to Chapters 1, 6, 7, 9 and 13 for specific details.

3.1.3 Protection and Reactivity Control Systems

3.1.3.1 Criterion 20 - Protection System Functions

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

RESPONSE

A fully automatic protection system with appropriate redundant channels is provided to cope with transients where insufficient time is available for manual corrective action. The design basis for all protection systems is in accordance with the intent of IEEE Standard 279-1971 and IEEE Standard 379-1972. The Reactor Protection System automatically initiates a reactor trip when any variable monitored by the system or combination of monitored variables exceed the normal operating range. Setpoints are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

Reactor trip is initiated by removing power to the rod drive mechanisms of all the full length Rod Cluster Control Assemblies. This causes the rods to insert by gravity rapidly reducing the reactor power output. The response and adequacy of the protection system has been verified by analysis of anticipated transients.

The Engineered Safety Features (ESF) Actuation System automatically initiates emergency core cooling, and other safeguards functions, by sensing accident conditions using redundant analog channels measuring diverse variables. Manual actuation of safeguards may be performed where ample time is available for operator action. The ESF Actuation System automatically trips the reactor on manual or automatic safety injection signal generation.

The Protection System is discussed in Chapter 7.

3.1.3.2 Criterion 21 - Protection System Reliability and Testability

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

RESPONSE

The Protection System is designed for high functional reliability and in-service testability. The Protection System meets the single failure requirements of IEEE Standard 279-1971 during normal operation. The rationale that justifies the exception to the single failure criterion during channel bypass for one-out-of-two systems is equally applicable to the two-out-of-three and two-out-of-four systems addressed in WCAP-10271. See Subsections 7.2.2 and 7.3.2 for a detailed discussion of compliance with this criterion.

3.1.3.3 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.

RESPONSE

Protection system components are designed and arranged so that the environment accompanying any emergency situation in which the components are required to function does not result in loss of the safety function. Various means are used to accomplish this. Functional diversity has been designed into the system. The extent of this functional diversity has been evaluated for a wide variety of postulated accidents. Diverse protection functions will automatically terminate an accident before intolerable consequences can occur. See Sections 7.2 and 7.3 for details.

High quality components, conservative design and applicable quality control, inspection, calibration, and tests are used to guard against common-mode failure. Qualification testing is performed on the various safety systems to demonstrate functional operation at normal and post-accident conditions of temperature, humidity, pressure, and radiation for specified periods, as required. The test results indicate no loss of the protection function. Refer to Sections 3.10 and 3.11 for further details.

3.1.3.4 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.

RESPONSE

The Protection System is designed with due consideration of the most probable failure modes of the components under various perturbations of the environment and energy sources. Each reactor trip channel is designed on the de-energize-to-trip principle so that loss of power, disconnection, open channel faults, and the majority of internal channel short-circuit faults cause the channel to go into its tripped mode. The Protection System is discussed in Sections 7.2 and 7.3.

3.1.3.5 Criterion 24 - Separation of Protection and Control Systems

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

RESPONSE

The Protection System is separate and distinct from the Control Systems. Control Systems may be dependent on the Protection System in that control signals are derived from protection system measurements, where applicable. These signals are transferred to the Control System by isolation amplifiers which are classified as protection components. The adequacy of system isolation has been verified by testing under conditions of postulated credible faults. The 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 System leaves intact a system which satisfies the requirements of the Protection System.

The Protection System is discussed in Chapter 7.

3.1.3.6 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.

RESPONSE

The Protection System is designed to limit reactivity transients so that design limits are not exceeded. Reactor shutdown by full length rod insertion is completely independent of the normal control function since the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. Thus in the postulated accidental withdrawal, (assumed to be initiated by a control malfunction) flux, temperature, pressure, level and flow signals would independently be generated. Any of these signals (trip demands) would operate the breakers to trip the reactor.

Analyses of the effects of possible malfunctions are discussed in Chapter 15. These analyses show that for postulated dilution during refueling, startup, or manual or automatic operation at power, the operator has ample time to determine the cause of dilution, terminate the source of dilution and initiate reboration before the shutdown margin is lost. The analyses show that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

The Protection System is discussed in Chapter 7.

3.1.3.7 Criterion 26 - Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design principals 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.

RESPONSE

Two Reactivity Control Systems are provided. These are Rod Cluster Control Assemblies (RCCAs) and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

During operation the shutdown rod banks are fully withdrawn. The full length Control Rod System automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks along with the full length control banks are designed to shutdown the reactor with adequate margin under conditions of normal operation and anticipated operational occurrences thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses and the most reactive rod cluster is assumed to be in the fully withdrawn position.

Sufficient boron can be provided to maintain the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Details of the construction of the RCCAs are presented in Chapter 4 and the operation is discussed in Chapter 7. The means of controlling the boric acid concentration is described in Chapter 9. Performance analyses under accident conditions are included in Chapter 15.

3.1.3.8 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.

RESPONSE

The facility is provided with means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. These means are discussed in detail in Chapters 4 and 9. Combined use of the Rod Cluster Control System and the Chemical Shim Control System permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. The single highest worth control cluster is assumed to be stuck full-out upon trip for this determination.

3.1.3.9 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 (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

RESPONSE

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to values that prevent rupture of the reactor coolant pressure boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The maximum positive reactivity insertion rates for the withdrawal of Rod Cluster Control Assemblies (RCCAs) and the dilution of the boric acid in the Reactor Coolant System are limited by the physical design characteristics of the RCCAs and the Chemical and Volume Control System. Technical Specifications on shutdown margin and on RCCA insertion limits and bank overlaps as functions of power provide additional assurance that the consequences of the postulated accidents are no more severe than those presented in the analyses of Chapter 15. Reactivity insertion rates, dilution and withdrawal limits are also discussed in Section 4.3. The capability of the Chemical and Volume Control System to avoid an inadvertent excessive rate of boron dilution is discussed in Chapter 15.

Assurance of core cooling capability following Condition IV accidents, such as rod ejection, steam line break, etc., is given by keeping the reactor coolant pressure boundary stresses within faulted condition limits, as specified by applicable ASME codes. Structural deformations are checked also, and limited to values that do not jeopardize the operation of necessary safety features.

3.1.3.10 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.

RESPONSE

The protection and reactivity control systems are designed to assure extremely high probability of performing their required safety functions in the event of anticipated operational occurrence. Likely failure modes of system components are designed to be safe modes. Equipment used in these systems is designed, constructed, operated and maintained with a high level of reliability. Loss of power to the Protection System results in a reactor trip. Details of system design are covered in Chapter 7. Also refer to the discussions of GDC-20 through 25.

3.1.4 Fluid Systems

3.1.4.1 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.

RESPONSE

Reactor coolant pressure boundary components are designed, fabricated, inspected and tested in conformance with ASME Boiler and Pressure Vessel Code Sec. III, Rules for Construction of Nuclear Power Plant Components. All components are classified according to ANSI N 18.2A-1975 and are accorded the quality measures appropriate to the classification. The design bases and evaluations of reactor coolant pressure boundary components are discussed in Chapter 5.

Leakage is detected by an increase in the amount of makeup water required to maintain a normal level in the pressurizer. The reactor vessel closure joint is provided with a temperature-monitored leak-off between double gaskets. Leakage into the reactor containment is collected in the containment building sump where it is monitored.

Leakage is also detected by measuring the airborne activity. Monitoring the inventory of rector coolant in the system at the pressurizer, volume control tank and coolant drain collection tanks make available an accurate assessment of integrated leakage.

The Reactor Coolant Pressure Boundary Leakage Detection System is discussed in Subsection 5.2.5.

3.1.4.2 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 (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual steady-state and transient stresses, and (4) size of flaws.

RESPONSE

Close control is maintained over material selection and fabrication for the Reactor Coolant System to assure that the boundary behaves in a non-brittle manner. The Reactor Coolant System materials which are exposed to the coolant are corrosion resistant stainless steel or Inconel. The reference temperature (RT_{NDT}) of the reactor vessel structural steel is established by Charpy V-notch and drop weight tests in accordance with 10 CFR 50, Appendix G.

As part of the reactor vessel specification, certain requirements which are not specified by the applicable ASME Codes are performed, as follows:

1. Ultrasonic Testing - In addition to code requirements, the performance of a 100 percent volumetric ultrasonic test of reactor vessel plate for shear wave and a post-hydro test ultrasonic map of all welds in the pressure vessel are required. Cladding bond ultrasonic inspection to more restrictive requirements than those specified in the code are also required to preclude interpretation problems during in-service inspection.

2. Radiation Surveillance Program - In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation and post-irradiation testing of Charpy V-notch and tensile specimens.

   These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the reference transition temperature approach and the fracture mechanics approach, and are in accordance with ASTM E185-79, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," and the requirements of 10 CFR 50, Appendix H.

3. Reactor vessel core region material chemistry (copper, phosphorous and vanadium) is controlled to reduce sensitivity to embrittlement due to irradiation over the life of the plant.

The fabrication and quality control techniques used in the fabrication of the Reactor Coolant System are equivalent to those used for the reactor vessel. The inspections of reactor vessel, pressurizer, piping, pumps, and steam generator are governed by ASME Code requirements. See Chapter 5 for details.

Allowable pressure-temperature relationships for plant heatup and cooldown rates are calculated using methods derived from the ASME Code, Section III, Appendix G, "Protection Against Non-Ductile Failure." The approach specifies that allowed stress intensity factors for all vessel operating conditions shall not exceed the reference stress intensity factor (KIR) for the metal temperature at any time. Operating specifications include conservative margins for predicted changes in the material reference temperatures (RT_{NDT}) due to irradiation.

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

RESPONSE

The design of the reactor coolant pressure boundary provides the capability for accessibility during service life to the entire internal surfaces of the reactor vessel, certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads, and external surfaces of the reactor coolant piping except for the area of pipe within the primary shielding concrete. The inspection capability complements the leakage detection systems in assessing the pressure boundary component's integrity. The reactor coolant pressure boundary will be periodically inspected under the provisions of ASME Boiler and Pressure Vessel Code, Section XI. Details of the in-service inspection programs are presented in Subsection 5.2.4.

Monitoring of changes in the fracture toughness properties of the reactor vessel core region plates, forging, weldments and associated heat treated zones are performed in accordance with 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." Samples of reactor vessel plate materials are retained and catalogued in case future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The observed shifts in RT_{NDT} of the core region materials with irradiation will be used to confirm the allowable limits calculated for all operational transients. See Section 5.3 for further details.

3.1.4.4 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.

RESPONSE

The Chemical and Volume Control System provides a means of reactor coolant makeup and adjustment of the boric acid concentration. Makeup is added automatically if the level in the volume control tank falls below the preset level. The high-pressure centrifugal charging pumps provided are capable of supplying the required makeup and reactor coolant seal injection flow when power is available from either onsite or offsite electric power systems. These pumps also serve as high head injection pumps. Functional reliability is assured by provision of standby components assuring a safe response to probable modes of failure. Details of system design are included in Sections 6.3 and 9.3, with details of the Electric Power System included in Chapter 8.

3.1.4.5 Criterion 34 - Residual Heat Removal

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the 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.

RESPONSE

The Residual Heat Removal System, in conjunction with the Steam and Power Conversion System, is designed to transfer the fission production decay heat and other residual heat from the reactor core within acceptable limits. Initiation of the residual heat removal system occurs at approximately 350°F and 425 psig.

Suitable redundancy at temperatures below 350°F is accomplished with two RHR drop lines, the two residual heat removal pumps (located in separate compartments with means available for draining and monitoring of leakage), the two heat exchangers and the associated piping, cabling and electric power sources. The Residual Heat Removal System is able to operate on either onsite or offsite electrical power systems.

Suitable redundancy at temperatures above approximately 350°F is provided by the four steam generators and attendant piping system.

Details of the system design are in Subsection 5.4.7 and Chapter 10, with details of the Electric Power System included in Chapter 8.

3.1.4.6 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 (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

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

RESPONSE

An emergency core cooling system is provided to cope with any loss-of-coolant accident in the plant design basis. Abundant cooling water is available in an emergency to transfer heat from the core at a rate sufficient to maintain the core in a coolable geometry, and to assure that clad metal-water reaction is limited to less than one percent. Adequate design provisions are made to assure performance of the required safety functions even with a single failure.

Details of the capability of the system are included in Section 6.3 with details of the Electric Power System included in Chapter 8. An evaluation of the adequacy of the system functions is included in Chapter 15. Performance evaluations are conducted in accordance with 10 CFR Part 50.46 and Appendix K.

3.1.4.7 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 integrity and capability of the system.

RESPONSE

Design provisions facilitate access to the critical parts of the injection piping and valves for visual inspection and for nondestructive inspection where such techniques are desirable and appropriate. The design is in accordance with ASME, Section XI requirements.

The components outside the containment are accessible for leak-tightness inspection during operation of the reactor.

Details of the inspection program for the Emergency Core Cooling System are included in Sections 6.3, 6.6 and Chapter 16.

3.1.4.8 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 (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the Protection System, the transfer between normal and emergency power sources, and the operation of the Associated Cooling Water System.

RESPONSE

Each active component of the Emergency Core Cooling System (ECCS) may be individually actuated on the normal power source or transferred to the emergency power source at any time during appropriate plant periodic tests.

Tests may be performed during shutdown to demonstrate proper automatic operation of the ECCS, and assure the structural and leak-tight integrity of the system components. The details of the ECCS testing program are included in Section 6.3 and Chapter 16. The emergency power system testing program is described in Chapter 8.

3.1.4.9 Criterion 38 - Containment Heat Removal

A system to remove heat from the reactor containment shall be provided. The system 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 acceptably low levels.

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

RESPONSE

A containment spray system is provided to remove heat from the reactor containment following an accident. Its function is to maintain the pressure and temperature of the containment below the design values at all times. The system consists of two independent identical subsystems supplied from separate power buses. No single failure causes loss of more than half of the installed 200 percent cooling capacity. The system is described in Subsection 6.2.2.

Electrical facilities are described in Chapter 8.

3.1.4.10 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.

RESPONSE

The Containment Heat Removal System is accessible for inspection, as described below:

a. The containment spray headers and nozzles are accessible for inspection when the reactor is shut down or at zero power.

b. The containment spray heat exchanger, spray pumps and associated valves are located in an enclosure adjacent to the reactor containment. These components are readily available for inspection at all times.

c. The containment recirculation sump strainer external surface is accessible for inspection. If necessary, the pump inlet plenum can be accessed using remote equipment when the reactor is shut down.

d. The refueling water storage and spray additive tank is readily accessible for inspection at all times.

For details, see Subsection 6.2.2.

3.1.4.11 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 (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the Protection System, the transfer between normal and emergency power sources, and the operation of the Associated Cooling Water System.

RESPONSE

System piping, valves, pumps, heat exchanger, and other components of the Containment Heat Removal Systems are arranged so that each component can be tested periodically for operability. The delivery capability of the Containment Spray System is tested periodically, to the extent practical, up to the last isolation valve before the spray nozzles. The delivery capability of the spray nozzles is tested periodically by blowing low-pressure air through the nozzles and verifying flow. The Containment Spray System is tested for operational sequence as close to the design as practical.

For details, see Chapter 6 for the testing of the Containment Heat Removal System; Chapter 8 for the testing of the Emergency Power System.

3.1.4.12 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 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 redundance 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.

RESPONSE

In the event of a loss-of-coolant accident, the Containment Spray System sprays a basic sodium hydroxide-borate solution into the containment atmosphere to remove fission product iodine. The spray system consists of two independent subsystems, each supplied from separate buses. Either subsystem alone can provide the iodine removal capacity for which credit is taken in Chapter 15.

The Post-Accident Hydrogen Control System is designed with redundant components so that a single failure does not prevent operation of the system. This system is described in Chapter 6.

A hydrogen mixing system is provided, comprised of redundant fans and ductwork to prevent the concentration of hydrogen. This system is described in Chapter 9.

The details of the Electric Power System are included in Chapter 8.

No single failure causes the subsystems to fail to function.

The secondary containment is maintained at a negative pressure, and acts as a collection system for containment leakage, which is collected and filtered prior to discharge to the environment.

3.1.4.13 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 systems.

RESPONSE

The Containment Atmosphere Cleanup Systems are designed and located so that they can be inspected periodically, as required. The spray headers and nozzles can be air-tested, as described in the response to Criterion 39.

The containment combustible gas control system components can be inspected and periodically tested.

See Chapter 6 for details.

3.1.4.14 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 leak-tight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves 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 of applicable portions of the Protection System, the transfer between normal and emergency power sources, and the operation of associated systems.

RESPONSE

The Containment Atmosphere Cleanup System can be tested as follows:

a. Operation of the spray pumps can be tested by recirculation to the refueling water storage tank through a test line.

b. System valves can be operated through their full travel.

c. Leak-tightness of the system is checked during testing.

See Chapter 6 for details.

Power transfer is described in Chapter 8.

3.1.4.15 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 electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operations (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

RESPONSE

The Cooling Water Systems for safety-related systems and components consist of the Primary Component Cooling Water System, the Service Water System and the ultimate heat sink. Each system has two trains that are capable of meeting the system safety function, assuming a single failure. See Chapter 9 for details. See Chapter 8 for electrical system details.

The Primary Component Cooling System is a closed system. During normal operation, it removes heat from various process system components, including letdown coolers and spent fuel pool coolers. This system also serves the residual heat removal and containment spray heat exchangers. Service water passes through the tube side of the primary component cooling water heat exchangers and transfers heat to the ultimate heat sink, which is either the Atlantic Ocean or a seismic Category 1 mechanical draft evaporative cooling tower.

3.1.4.16 Criterion 45 - Inspection of Cooling Water System

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

RESPONSE

Important components of cooling water systems, such as pumps, strainers, heat exchangers and valves are located in accessible areas. These components have suitable manholes, handholes, inspection ports or other appropriate design and layout features to allow periodic inspection. In the case of buried piping, integrity will be verified by pressure testing.

See Chapter 9 and 10 for details.

3.1.4.17 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 leak-tight integrity of it 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.

RESPONSE

The design provides for periodic testing of active components of the Cooling Water System for operability and functional performance as well as assuring the structural and leak-tight integrity of its components. Preoperational performance tests of the components are made in the manufacturer's shop. An initial system flow test demonstrates proper functioning of the system. Thereafter, periodic tests ensure that components are functioning properly.

Each active component of the Cooling Water System may be individually connected to the preferred power source at any time during reactor operation to demonstrate operability. Many active components are operating during normal operation, thereby demonstrating their availability. Remote operated valves may be exercised and actuation circuits tested. The automatic actuation circuitry, valves, and pump breakers also may be checked when integrated system tests are performed during a planned cooldown of the Reactor Coolant System.

See Chapter 9 for details. See Chapter 8 for electrical system testing details.

3.1.5 Reactor Containment

3.1.5.1 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 consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

RESPONSE

The design of the containment is based on two containment design basis accidents. One assumes a double-ended rupture of the largest reactor coolant pipe (LOCA); the other the rupture of a main steam line inside containment. The maximum calculated atmospheric pressure and temperature reached within the containment during the LOCA are 49.6 psig and 273°F; the maximum atmospheric pressure and temperature attained during a main steam line rupture is 36.1 psig and 364°F. A containment design pressure of 52.0 psig has been selected to provide ample margin to allow for increased energy sources. The peak liner temperature following a LOCA is calculated to be less than the design temperature of 271°F. Although the containment atmospheric temperature following a MSLB is higher than that following a LOCA, the containment liner temperature will not exceed 271°F, since a lower heat transfer coefficient will result under the super-heated atmospheric condition during the MSLB.

See Subsection 3.8.1 for containment loading combinations and Subsection 6.2.1 for design evaluation.

The containment electrical penetrations are designed so that the containment structure can, without exceeding the design leakage rate, accommodate the calculated pressure, temperature and other environmental conditions resulting from any loss-of-coolant accident. See Section 8.3 for discussion of containment electrical penetrations and protections of containment electrical penetrations.

3.1.5.2 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 (1) its ferritic materials behave in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

RESPONSE

The reactor containment boundary is designed with sufficient margins to meet the requirements of Criterion 51.

All ferritic materials used for the fabrication of the containment liner and hatches are in accordance with ASME Boiler and Pressure Vessel Code, Section III, Divisions 2 and 1, respectively.

For further details, refer to the following subsections:

a. Concrete Containment 3.8.1

b. Steel Containment (Hatches) 3.8.2

c. Containment Functional Design 6.2.1

3.1.5.3 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.

RESPONSE

The containment system is designed and constructed and the necessary equipment is provided to permit periodic integrated leakage rate tests during plant lifetime, in accordance with reduced pressure test program requirements of Appendix J of 10 CFR 50.

Details concerning the conduct of periodic integrated leakage rate tests are presented in Chapter 6.

3.1.5.4 Criterion 53 - Provisions for Containment Testing and Inspection

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

RESPONSE

There are special provisions for conducting individual leakage rate tests on applicable penetrations. Penetrations are visually inspected and pressure-tested for leak-tightness at periodic intervals, as required by Appendix J of 10 CFR 50.

Refer to Chapter 6 for details.

3.1.5.5 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 that 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.

RESPONSE

Piping systems penetrating primary reactor containment are provided with containment isolation valves, as described in responses to Criteria 55, 56 and 57.

Capability is provided for periodic testing of isolation valves during normal operation or during shutdown conditions in accordance with ASME Section XI, Article IWV.

Isolation valve leak test provisions are provided for compliance with 10 CFR 50, Appendix J.

See Chapter 6 for details.

3.1.5.6 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 in-service inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

RESPONSE

Each line that is part of the reactor coolant pressure boundary that penetrates the containment is provided with isolation valves or barriers meeting this criterion. Instrument lines are designed in accordance with NRC Regulatory Guides 1.11, 1.141 and 1.151. Details and clarifications are provided in Subsection 6.2.4, 7.1.2.2a, and 7.3.1.1b.

3.1.5.7 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.

RESPONSE

Each line that connects directly to the containment atmosphere and penetrates containment is provided with containment isolation valves or barriers meeting this criterion, except where it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable. Instrument lines are designed in accordance with NRC Regulatory Guides 1.11, 1.141, and 1.151.

Details are given in Subsections 6.2.4, 7.1.2.2a, and 7.3.1.1b.

3.1.5.8 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.

RESPONSE

Each line that penetrates containment and is not connected directly to the containment atmosphere and is not part of the reactor coolant pressure boundary has at least one isolation valve outside containment near the penetration, which meets this criterion. Details are provided in Subsection 6.2.4.

3.1.6 Fuel and Radioactivity Control

3.1.6.1 Criterion 60 - Control of Releases of Radioactive Materials to the Environment

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

RESPONSE

Waste handling systems are incorporated in the facility design for processing and/or retention of normal operation radioactive wastes with appropriate controls and monitors to assure that releases do not exceed the guidelines of 10 CFR 20 and 10 CFR 50. The facility is also designed with provisions to prevent radioactivity release during accidents from exceeding limits of 10 CFR 100.

The Radioactive Waste Processing System, the design criteria, and amounts of estimated releases of radioactive effluents to the environment are described in Chapter 11.

3.1.6.2 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 system, (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.

RESPONSE

The Spent Fuel Pool and Cooling System, Fuel Handling System, Radioactive Waste Processing Systems, and other systems that contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions.

Components are designed and located so that appropriate periodic inspection and testing may be performed.

All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation dose rates and occupancy as discussed in Chapter 12.

Individual components that contain significant radioactivity are in confined areas adequately ventilated through appropriate filtering systems.

The Spent Fuel Cooling System provides cooling to remove residual heat from the fuel stored in the spent fuel pool. The system is designed with redundancy and testability to assure continued heat removal. The spent fuel cooling system is described in Chapter 9.

The spent fuel pool is designed so that no postulated accident could cause an excessive loss of coolant inventory.

3.1.6.3 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.

RESPONSE

Criticality in fuel storage areas is prevented by both physical separation of fuel assemblies and the presence of borated water in the fuel storage pool. The fuel storage racks are constructed so that the fuel assemblies may be inserted in prescribed locations only. These racks are designed with neutron absorbing material as an integral rack component to ensure sub-criticality even if assemblies are immersed in unborated water. Criticality prevention and criticality considerations are discussed in Sections 9.1 and 4.3, respectively.

3.1.6.4 Criterion 63 - Monitoring Fuel and Waste Storage

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

RESPONSE

Monitoring systems are provided to alarm when excessive temperatures or low water level occurs in the spent fuel pool. Appropriate safety actions are initiated by operator action.

Radiation monitors and alarms are provided as required to warn personnel of impending excessive levels of radiation.

The ventilation system in the Fuel Storage Building is operated in the fuel handling mode anytime irradiated fuel not in a cask in handled. In this operating mode, the building is maintained at a negative pressure and all exhaust air is filtered by charcoal filters prior to discharging the air to the atmosphere via the plant unit vent. In the unlikely event of a fuel handling accident, the filtration system is already operational and available to filter exhaust air from the building. See Chapter 9 for details.

The Radiation Monitoring System is described in Chapter 12.

3.1.6.5 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 for normal operations, including anticipated operational occurrences, and from postulated accidents.

RESPONSE

The containment atmosphere is continually monitored during normal and transient station operations, using the containment air particulate and radio-gas radiation monitors. In accident conditions, samples of the containment atmosphere provide data on existing airborne radioactive concentrations within the containment. Radioactivity levels contained within the facility gaseous and liquid effluent discharge paths and in the station environs are continually monitored during normal and accident conditions by the station radiation monitoring as described in Chapter 11 and 12.

In addition to the installed detection system, an environment radiation surveillance program monitors the exposure pathways to Man within a 5-mile radius from the station site.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 Seismic Classification

Seabrook Station structures, systems and components important to safety, as well as their foundations and supports, have been designed to withstand the effects of an Operating Basis Earthquake (OBE) and a Safe Shutdown Earthquake (SSE), and are thus designated as seismic Category I. These plant features are those 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 shutdown condition, or

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

Those structures, systems and components classified as seismic Category I are listed in Table 3.2-1 and Table 3.2-2. These structures, systems and components are designed to withstand seismic loads as discussed in Section 3.7.

Seismic Category I structures are sufficiently isolated or protected for non-seismic Category I structures to ensure that their integrity is maintained at all times. None of the plant structures is classified as partially seismic Category I, with the exception of the Circulating Water Pumphouse and the Waste Processing Building, which are described in Section 3.8. Several non-seismic Category I structures are designed against collapse onto seismic Category I structures due to SSE loadings. These are discussed in Subsection 3.7(B).2.

Where only portions of systems are classified as seismic Category I, the boundaries of the seismic Category I portions of the system extend to the first restraint beyond the isolation valves that isolate the part which is Category I from the nonseismic portion of the system.

Equipment and components that are not classified as seismic Category I, and whose collapse or failure could result in the loss of safety function of seismic Category I structures, systems or components, are checked to confirm their structural integrity against collapse or failure due to SSE loadings.

Where the collapse on non-Category I components would adversely affect the performance of Category I systems or components, they have been supported to withstand seismic loadings or isolated from the Category I systems or components by boundary restraints. The locations of the boundary restraints are shown on the pipe support location drawings and the pipe support detail drawings.

Where the collapse of non-Category I HVAC system components or ductwork would damage the other Category I system or components in the vicinity, they have been supported to withstand seismic loadings.

The seismic classifications presented in Table 3.2-1 and Table 3.2-2 are consistent with the recommendations of NRC Regulatory Guide 1.29.

3.2.2 Quality Group Classification

The system quality group classification applies to fluid systems, or portions of fluid systems, which are directly depended 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 maintenance in the safe shutdown condition

c. Contain radioactive material.

Fluid system components important to safety are classified in accordance with the ANSI N18.2a-1975, "Revision and Addendum to Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." This classification system is compatible with the requirements of NRC Regulatory Guide 1.26, and is submitted as an alternate acceptable method of meeting the intent of Regulatory Guide 1.26.

3.2.2.1 Safety Class Definitions

Components are classified as Safety Class 1, Safety Class 2, Safety Class 3, and nonnuclear-safety in accordance with their importance to nuclear safety. This importance, as established by the assigned safety class, is applied in the design, materials, manufacture or fabrication, assembly, erection, construction, and operation. A single system may have components in more than one safety class.

The definitions of safety classes listed apply to fluid pressure boundary components and the reactor containment. Supports that have a nuclear safety function are of the same safety class as the components that they support.

a. Safety Class 1

Safety Class 1 applies to components whose failure could cause a Condition III or Condition IV loss of reactor coolant. Condition III occurrences include incidents any one of which may occur during the lifetime of a plant. Condition IV occurrences are faults that are not expected to occur but are postulated because their consequences would include the potential for release of significant amounts of radioactivity. Condition IV faults are the most drastic which must be designed against, and thus represent the limiting design case.

b. Safety Class 2

Safety Class 2 applies to the reactor containment and to the following components:

1. Components of the reactor coolant pressure boundary not in Safety Class 1.

2. Components of safety systems that are necessary to remove heat directly from the reactor containment, to circulate reactor coolant for any safety system purpose, to control within the reactor containment radioactivity released, or to control hydrogen in the reactor containment. A safety system (in this context) is any system that is necessary to shut down the reactor, cool the core or cool another safety system or the reactor containment (after an accident), or it is any system that contains, controls, or reduces radioactivity released in an accident. Only those portions of a system that are designed primarily to accomplish one of those functions, or the failure of which could prevent accomplishing one of those functions, are included.

c. Safety Class 3

Safety Class 3 is applied to those components not in Safety Class 1 or Safety Class 2, the failure of which would result in release to the environment of radioactive gases normally required to be held for decay, or that are necessary to:

1. Provide or support any safety system function,

2. Control outside the reactor containment airborne radioactivity released in an accident, or

3. Remove decay heat from spent fuel.

d. Nonnuclear Safety

Nonnuclear safety applies to portions of the nuclear power plant not covered by Safety Classes 1, 2, or 3 that can influence safe, normal operation or that may contain radioactive fluids. Design of nonnuclear safety components shall be to applicable industry codes and standards.

3.2.2.2 Component Classification and Code Requirements

Fluid system component safety class designations are listed in Table 3.2-2. Table 3.2-2 also indicates industry codes and standards applicable to pressure-retaining components and associated safety systems, and provides other pertinent information. The safety classification and design standards for instrument sensing lines follow the guidelines of Regulatory Guide 1.151 with exceptions as listed in Subsection 7.1.2.12. The safety class designation boundaries of safety-related systems are shown on the piping and instrumentation diagrams in the referenced sections of the Updated FSAR.

HVAC system component safety class designations are listed in Table 3.2-4. Table 3.2-4 also indicates design criteria and guidelines used for ductwork and supports. The safety class designation boundaries of safety-related systems are shown on air flow diagrams and ductwork drawings.

Quality standards, as specified by Code classifications, are correlated to safety class designations in Table 3.2-3. Quality assurance programs applied to safety-related systems and components, and developed to meet the intent of 10 CFR 50 Appendix B, are presented in Chapter 17.

3.3 WIND AND TORNADO LOADINGS

3.3.1 Wind Loadings

The design of seismic Category I structures for normal wind loading is based on two criteria: (1) ASCE Paper No. 3269, "Wind Forces on Structures," Reference 1 (Subsection 3.3.3) and (2) ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures," Reference 2 (Subsection 3.3.3). The ASCE paper is used to derive the wind loading for the Containment Enclosure Building, while the ANSI guideline is used for wind loading on all other seismic Category I structures.

3.3.1.1 Design Wind Velocity

The design wind velocity at 30 feet above ground for the 100 year period of recurrence is 110 mph (See Subsection 2.3.1).

a. ASCE - The vertical wind velocity profile is interpolated from Table 1(b) of Reference 1 (Subsection 3.3.3). A gust factor of 1.1 is used in deriving wind loading based on guidelines presented in Reference 1 (Subsection 3.3.3).

b. ANSI - The vertical velocity profile and applicable gust factors are included in the velocity pressure profiles discussed in Subsection 3.3.1.2b.

3.3.1.2 Determination of Applied Forces

Wind loads are applied as uniform static loads on the horizontal and vertical projected areas of the structure walls and roof. Shielding effects provided by other structures are neglected unless a portion of the exposed surface is immediately adjacent to another structure. Only dead loads are considered in resisting uplift.

a. ASCE - The design wind velocities are converted to wind pressures by the following formula:

$$p = 0.00256(GV)^2 C_p$$

where:

p = design wind pressure (psf)
G = gust factor
V = wind velocity (mph)
C_p = pressure coefficient (Table 4, Reference 1)

The vertical profile of wind pressure on the Containment Enclosure Building is summarized in Table 3.3-1. The distribution of effective pressure coefficients, (C_p), for cylindrical and spherical structures is shown on Figure 3.3-1.

b. ANSI - The effective velocity pressure vertical profile, including gust factors for structures (q_F), parts of structures (q_p), and internal pressure (q_M) are taken from Tables 5, 6 and 12 (Exposure C) respectively of Reference 2. The effective velocity pressures are summarized in Table 3.3-2. The average pressure acting on a structural element is as follows:

$p = q\,C_p - q_M\,C_{pi}$

where:

p = average wind pressure (psf)

q = q_F or q_p whichever is appropriate

C_p = external pressure coefficient

q_M = internal velocity pressure

C_{pi} = internal pressure coefficient

Load combinations involving normal wind (discussed in Subsections 3.8.4.3 and 3.8.5.3) do not control the design of seismic Category I structures.

3.3.2 Tornado Loadings

3.3.2.1 Applicable Design Parameters

a. Tornado wind loads on seismic Category I structures are based on a single vortex tornado model which results in a maximum tangential velocity of 290 mph and a maximum translational velocity of 70 mph. The total maximum wind velocity is 360 mph assumed constant with respect to height. The pressure differential is 3 psi at a rate of 2 psi/second. See Subsection 2.3.1.2 for a discussion of the design basis tornado parameters.

b. Protection of equipment and systems located in the "closed" structures listed in paragraph c. below, is accomplished by using protective barriers consisting of dampers complete with the necessary accessories and instrumentation. These dampers are provided at inlet openings located in the exterior walls of the structures which provide outdoor air for the plant HVAC systems. These dampers are also provided at the openings in the exterior walls of the structures, through which air is exhausted from the various plant HVAC systems to the outside atmosphere. The dampers automatically close to avoid damage which could be caused by tornado induced depressurization within the structures to the systems and components located inside the structures. Ductwork and backdraft dampers located internal to the tornado dampers were evaluated for the positive tornado wind pressure loads considering the tornado dampers open.

Seismic Category I systems and components which are located in "open" structures are protected from tornado wind pressure loads and tornado-generated missiles. Such equipment and interior structures have been shown to be capable of performing their design function when subjected to tornado depressurization effects.

c. Of the seismic Category I structures described in Subsection 3.8.4.1, the following structures are protected from tornado depressurization effects and are considered "closed" structures.

1. Containment Enclosure Ventilation Area (including Containment)
2. Control Building
3. Electrical Tunnels
4. Emergency Feedwater Pump Building
5. Primary Auxiliary Building (including equipment vault areas)
6. Service Water Pumphouse (Switchgear Room Only)

3.3.2.2 Determination of Forces on Structures

a. The total wind velocity is transformed into the total tornado wind load on seismic Category I structures by means of either the ASCE Paper 3269, "Wind Forces on Structures" (Reference 1), or the ANSI Standard A58.1-1972, "Building Code Requirements for Minimum Design Loads in Building and Other Structures" (Reference 2). The ASCE Paper is used for the Containment Enclosure Building and the ANSI Standard is used for all other seismic Category I structures.

Velocity pressure is assumed to be constant with height. Maximum velocity pressure is based on the maximum tornado wind velocity and is assumed to occur at the radius of the tornado funnel at which the maximum velocity occurs.

The maximum velocity pressure, q_{max}, is given by the following formula:

$q_{max} = 0.00256V^2$

where, V = total tornado wind velocity - 360 mph

therefore, q_{max} = 332 psf

$q_{avg} = q_{max}\ C_s$

where,

q_{avg} = Average velocity pressure

C_s = Size factor depending on L (See Figure 3.3-2)

L = Load distribution length equal to:

1. the plan distance perpendicular to the wind over which the wind load can be distributed, or

2. the mean horizontal extent of the tributary area perpendicular to the direction of wind.

The total pressure (W_w) acting on a structure or structural element is equal to the external tornado wind pressure (W_{w1}) plus the internal wind pressure (W_{w2}). Calculation of the two pressure components (W_{w1} and W_{w2}) for the two controlling criteria (ASCE and ANSI) is shown in Table 3.3-3.

Variation of tangential velocity with radial distance from the tornado core is determined as follows:

$$V_t = \frac{r}{r_m} \times V_t \max \text{ for } 0 < r < r_m$$

$$V_t = \frac{r_m}{r_r} \times V_t \max \text{ for } r_m < r < r_{75}$$

where

V_t = tangential velocity at radius r

$V_{t\,max}$ = maximum tangential velocity (290 mph)

r = radius from centerline of tornado

r_m = radius of maximum tangential velocity (150 ft)

r_{75} = radius at which tangential velocity equals 75 mph (580 ft)

b. Venting of structures is not adopted as a means of transforming the tornado-generated differential pressure into an effective reduced pressure. The exterior walls and roof slabs of seismic Category I structures are subjected to the full 3 psi pressure drop.

c. The postulated horizontal and vertical tornado-generated missiles, and procedures used to design the structures or barriers against the effects of these missiles are described in detail in Subsections 3.5.1.4 and 3.5.3, respectively.

d. The effective wind loadings are combined in the following manner in order to determine the most adverse tornado effect on seismic Category I structures.

(I) $W_t = W_{w1} + W_{w2}$

(II) $W_t = W_p$

(III) $W_t = W_m$

(IV) $W_t = W_{w1} + 0.5W_p$

(V) $W_t = W_{w1} + W_{w2} + W_m$

(VI) $W_t = W_{w1} + 0.5W_p + W_m$

where, W_t = Total tornado load

W_{w1} = Tornado wind external pressure load

W_{w2} = Tornado wind internal pressure load

W_p = Tornado differential pressure drop = 3 psi (432 psf)

W_m = Tornado missile load

The most severe W_t of the above combinations is used in the seismic Category I load combinations discussed in Subsections 3.8.1 and 3.8.4.

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

Any non-Category I structure, which as a result of a catastrophic failure could fall upon a seismic Category I structure and, thereby, effect the safety function of the structure, is designed for tornado wind in the following manner:

a. Exterior siding, nonstructural walls and partitions, and the roof slab are considered expendable and can be permitted to fail during a tornado. The failure of these elements does not generate missiles of greater severity than those listed in Subsection 3.5.1.4.

b. The structural frame is checked for tornado wind pressure by assuming that the wind load acts on the wall fully sided, or on one-third of the siding area and the remainder of the exposed steel frame including girts, crane girders, etc., whichever results in the most severe loads. The resulting tornado wind loading is used in the load combination for seismic Category I structures (other than containment), and the stresses checked against the appropriate acceptance criteria. (See Subsection 3.8.4.)

The nonseismic Category I structures which are designed for tornado wind using the procedure outlined above are summarized in Table 3.3-4.

The large roll-up steel door located in the east wall of the seismic Category I Fuel Storage Building is not designed for tornado loadings. Its failure will not result in any adverse consequences affecting the safety function of the Fuel Storage Building or adjoining structures.

3.3.3 References

1. ASCE Paper No. 3269, "Wind Forces on Structures," Transactions, Vol. 126, Part II, 1961, P1124.

2. ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Others Structures," American National Standards Institute, New York, 1972.

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 Flood Protection

3.4.1.1 Flood Protection Measures for Seismic Category I Structures

All safety-related systems and components, as identified in Table 3.2-1 and Table 3.2-2, are protected against floods. These systems and components are protected by the structures which house them, and/or by being located above a maximum water level not exceeding 21 feet MSL, postulated to result during the combined PMH-SPS event.

Seismic Category I structures house safety-related equipment, and are listed in Table 3.2-1. These structures are designed to withstand a depth of still water of not exceeding 21 feet MSL. The locations of the walls of those structures which could be subject to wave runup to elevation of 21.8 feet MSL are shown in Figure 2.4-21.

The only access openings in any exterior wall that are below the design flood level are the rolling steel door in the Fuel Storage Building (west wall) and a personnel emergency exit (east wall), located at elevation 20 feet 6 inches MSL (See Figure 1.2-16), and the double doors into the entrance vestibule of the Equipment Vault section of the Primary Auxiliary Building, located at elevation 20 feet 8 inches MSL (See Figure 1.2-13). Flood protection for the Fuel Storage Building is provided by a curb at elevation 21.5 feet MSL located on column line 3. This door is closed during normal plant operation, thus providing the same protection against wave run-up as the other vertical building walls. The floor of the vestibule into the Equipment Vault section of the Primary Auxiliary Building is sloped up 4 inches so that the high point in the floor is at elevation 21 feet MSL.

To minimize potential in-leakage from such phenomena as minute cracks in structure walls or leakage waterstops, all below-grade safety-related structures, other than the pumphouse, cooling tower, electrical duct banks and manholes are waterproofed on the exterior face. Such cracks will be minimal because of the structures being heavily reinforced due to the various design criteria. Typical details of waterproofing for penetrations are shown in Figure 3.4-1. In addition, sump pumps are provided in all seismic Category I structures other than Category I manholes where seepage that could occur could affect safety-related equipment. All structures except the Control Building, which is above grade, are protected in this manner. The pipe chases below the Control Building drain into the Emergency Feedwater Pump Building, which is protected by sump pumps.

Because all safety-related structures, systems and components are provided with protection against floods, as described in the preceding paragraphs, it will not be necessary to bring the reactor to a cold shutdown for the most severe flood anticipated at the plant.

Protection against possible flooding from liquid carrying systems due to pipe rupture or fire protection activities is discussed in the following sections:

Section 3.6	Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping
Subsection 9.3.3	Equipment and Floor Drainage System
Subsection 9.5.1	Fire Protection System

3.4.1.2 Plant Dewatering System

Seabrook Station was not originally designed with a dewatering system, because it was believed that the in-leakage prevention methods described in 3.4.1.1 would be adequate to prevent water ingress. Over the years, it has become evident that the mitigation methods were not completely effective at preventing in-leakage. The presence of groundwater in-leakage is a plant concern from the perspectives of:

- Housekeeping (i.e., maintaining walking surfaces in lower elevations of building structures, below grade, dry) is difficult because the in-Leakage is continuous
- In-leakage occurs in some areas where it could spread (tritium) contamination (i.e., a small wet area becomes a larger wet area)
- In-leakage allows communication of certain plant drainage systems (e.g., the spent fuel pool leak off system) with the soil and groundwater beneath the buildings.

A plant dewatering system has been installed which can further mitigate in-leakage of groundwater in the lower elevations of the plant. The purpose is to routinely pump water from beneath the plant structures, to reduce the static hydraulic head outside the building concrete and reduce the in-leakage. This allows the original mitigative measures to function properly. A pump is installed in the existing well at (+)7’ elevation of the PAB. This pump discharges the water to the roof drain system, which then flows to the storm drain system and out to circulating water for discharge. Routine monitoring of this flowpath is performed per station operating procedures.

Existing pipe penetrations located in EF103/EFST1 have been utilized as a groundwater low point. These penetrations have been directed to a nearby sump (1-DF-5). This sump discharges to the existing plant storm drainage system.

A pump is installed in the RHR Vault "B" stairwell at (-)61' elevation of the Equipment Vault. This pump discharges the water to the roof drain system, which then flows to the storm drain system and out to circulating water for discharge. Routine monitoring of this flow path is performed per station operating procedures.

A pump is installed in the containment annulus at –32' elevation. This pump discharges the ground water in the containment annulus to the roof drain system. The connection to the drain system is installed at 240 degrees azimuth of the containment annulus. Routine monitoring of this flow path is performed per station operating procedures.

A ground water collection tank and pump are installed in the "B" Electrical Tunnel, West stairwell at (-) 20'-0" elevation. The pump discharges the water to the turbine building roof drain system, which then flows to the storm drain system and out to circulating water for discharge via the outfall. Routine monitoring of this flow path is performed per station operating procedure. A 50.59 Applicability and Screening are performed to support this change.

A pump has been installed in a pre-existing, construction-era French drain well that terminates just north of the EFW Pumphouse and draws from outside the eastern periphery of the Containment Building extending as far south as the Fuel Storage Building. The pump discharges the water to the storm drain system and then to circulating water for discharge via that outfall. This well, not directly below plant structures, will withdraw water from underneath them. Routine monitoring of this flow path is performed per station operating procedures.

3.4.2 Analytical and Test Procedures

The methods and procedures by which the static and dynamic effects of the design basis flood conditions or design basis groundwater conditions identified in Section 2.4 are applied to safety-related structures, systems and components are described in Subsections 3.8.1, 3.8.4 and 3.8.5. In the design of seismic Category I structures, the groundwater is assumed to be at 20 feet MSL, the plant grade. As discussed in Subsection 3.4.1, the design basis flood could cause a depth of stillwater on the plant grade of not exceeding 1 foot, increasing the top of the hydrostatic head to elevation not exceeding 21 feet MSL.

Dynamic effects of the design basis flood were considered, but found to be negligible. The maximum depth of stillwater is not exceeding one foot above plant grade, and the maximum wave runup in local regions is 1.8 feet above plant grade. Any dynamic effects produced by these occurrences were evaluated and found to be negligible and, due to the relatively large masses of the reinforced concrete structures, can be neglected.

The coincident wind loadings used in the design of the structures are those associated with the Probable Maximum Hurricane (PMH), during which the design basis flood occurs. These wind loading are discussed in Subsection 3.3.1.

The static and dynamic effects on foundations due to the design basis flood are considered in the design of all seismic Category I structures. Dynamic forces due to wave runup are negligible compared to the forces that are considered because of other design criteria.

All seismic Category I structures are designed to prevent uplift or overturning, considering the effects of buoyant forces along with other forces (see Subsection 3.8.5).

3.5 MISSILE PROTECTION

Structures, shields and barriers are provided as protection against the effects of both internally and externally generated missiles, an exception being the turbine-generated missiles discussed in Subsection 3.5.1.3.

Structures that have been analyzed for missile damage protection capabilities are those that:

- House or service safety-related systems and components required for the safe shutdown of the reactor and its maintenance in a safe shutdown condition;
- If damaged, could result in a significant uncontrolled release of radioactivity.

3.5.1 Missile Selection and Description

3.5.1.1 Internally-Generated Missiles (Outside Containment)

The principal design basis for missiles generated outside containment, but internal to the plant site, is that such missiles shall not cause loss of function of any design feature provided for either continued safe operation or shutdown of the reactor during operating conditions, operational transients, and postulated accident conditions associated with the effects of missile formation.

Two general sources of internally generated missiles outside containment which have been considered are:

a. Those due to rotating component failures

b. Those due to pressurized component failures.

Components evaluated for possible missile generation outside containment, and discussed below, are valves in high pressure systems, temperature sensing element wells, pumps, and motor generator sets.

Catastrophic failure of major components, e.g., pumps, heat exchangers, etc., is not considered credible because of: (a) material characteristics, inspections, quality control during fabrication, erection, and operation; (b) the maximum no-load speed of the components is the operating speed of their motors; (c) fans are tested to 125 percent of operating speed; and (d) where necessary internally generated missile analysis will be conducted to verify that no such missiles can degrade the adjoining safety-related equipment.

a. Valves

1. Valve stems are not considered to be credible sources of missiles, since all valves in high energy lines have stems with a back seat that effectively eliminates the possibility of valve stems being ejected even if the stem threads fail. Analysis has indicated that the back seat or the upset end will not penetrate the bonnet. Additional interference is encountered with air and motor-operated valves.

2. Valves with a nominal diameter larger than 2 inches have been designed against bonnet-body connection failure and subsequent bonnet ejection by:

    (a) Using the design practice of ASME Section III which limits the allowable stress of bolting material to 25 percent of minimum yield strength

    (b) Using the design practice of ASME Section III for flange design

    (c) Controlling the load during the bonnet-body connection stud tightening process.

3. Valves with nominal diameter of 2 inches or smaller are forged and have screwed or bolted bonnets. Bonnet fastening systems are designed to code requirements. The pressure-containing parts are designed per criteria established by ASME Section III code specification.

4. The pressure-containing parts of valves, including the flanges and studs, are designed to the requirements of the ASME Code for safety class valves and ANSI B16.34 and B16.5 for non-nuclear safety class valves.

5. The proper stud torquing and tightening procedures, and the use of torque wrenches, with indication of the applied torque or stud tension, limit the stress of the studs to the allow-able limits established in the ASME Code. This stress level is far below the material yield.

6. The complete valves are hydrotested per code requirements.

7. Suitable nondestructive tests are performed, as appropriate, in accordance with code requirements.

It is, therefore, concluded that there are no credible sources of missiles associated with valves located outside containment.

b. Temperature Sensing Element Wells

Temperature sensing element wells in safety-related system piping outside containment are designed, manufactured and welded to the same codes as the piping in which they are located and are, therefore, also not considered credible missiles. Where possible, the sensing element well is located so that, if in the remote case it did become a missile, it would impact directly on a nearby reinforced concrete wall, floor or ceiling. These barriers are designed to withstand such missiles. Where required, a local missile barrier will be designed and installed in individual cases.

c. Pumps

1. Motor-Driven

Pumps located outside containment have been evaluated for missiles associated with overspeed failure. The maximum no-load speed for these pumps is equivalent to the operating speed of their motors. Consequently, no pipe break or single failure in the suction line would increase pump speed over that of the no-load condition. Furthermore, there are no pipe break plus single failure combinations which could result in a significant increase in pump suction or discharge head. Therefore, no overspeed is expected, and missiles associated with pumps outside containment are not credible.

2. Turbine-Driven

The turbine-driven Emergency Feedwater (EFW) pump is identical in size and design to the motor-driven EFW pump, and operates at the same speed. The turbine-driven EFW pump is not considered to be a credible missile source for the same reasons stated above for the motor-driven pump.

The turbine unit is equipped with both a speed-limiting governor and an overspeed limiting trip. The Speed Governing System is designed to assure rapid controlled acceleration without overspeeding. The overspeed governor consists of a mechanical pin-type device which trips shut the turbine steam inlet valve at 125 percent overspeed. Repeatability accuracy of this trip is within ± 50 rpm.

The turbine itself is of a solid wheel, single stage design and is not considered a credible missile source. This unit has been designed to start or operate on 100 percent water as well as being able to withstand the severe punishment of intermittent water slugs. During testing of similar type units, water slugs were injected into the turbine while it was operating normally. These water slugs ranged from 50 to 600 gallons. Following these tests, detailed examinations confirmed that the turbine sustained no wear deformation or damage.

Further assurance of the integrity of the turbine drive unit is provided by the Quality Assurance program of the vendor. During manufacturing, the vendor follows the intent of ASME Section III standards, Appendix B to 10 CFR 50 (Quality Assurance Criteria Requirements of the Code of Federal Regulations) and ANSI 45.2. Certified material is used on all major components so that complete traceability is possible. Welders are qualified to ASME Section IX standards and non-destructive test personnel are qualified to SNT standards. The integrity of every turbine casing is confirmed by 100 percent mag-particle testing. Every high pressure component is subjected to a thorough X-ray analysis. Every shaft, as well as every wheel is ultrasonically tested. Additionally, conversations with the turbine manufacturer have indicated that for the turbine wheel to separate, speeds in excess of 14,000 RPM would be necessary. This is approximately a 400 percent overspeed for this unit.

The motor-driven EFW pump is oriented perpendicular to the turbine-driven EFW pump so that in the unlikely event that pump or turbine missiles are generated the other pump will not be affected.

d. Motor-Generator Sets

The fabrication specifications of the motor-generator set flywheels control the material to meet ASTM A533-7D, Grade B, Class 1, with inspections per MIL-I-45208A and flame cutting and machining operations governed to prevent flaws in the material. Nondestructive testing for nilductility (ASTM E208), Charpy V-notch (ASTM A593), ultrasonic (ASTM A577 and A578) and magnetic particles (ASME Section III, NB2545) is performed on each flywheel material lot. In addition to these requirements, stress calculations are performed consistent with guidelines of the ASME Code Section III, Appendix A, to show that the combined primary stresses due to centrifugal forces and the shaft interference fit will not exceed one-third of yield strength at normal operating speeds (1800 rpm), and will not exceed two-thirds of the yield strength at 25 percent overspeed. However, no overspeed is expected because the flywheel weighs approximately 1300 pounds and has dimensions of 35.26 inches in diameter by 4.76 inches wide. The flywheel mounted on the generator shaft, which is directly coupled to the motor shaft, is driven by a 200 hp, 1800 rpm induction motor. The torque developed by the motor is insufficient for overspeed. Therefore, there are no credible missiles from the motor-generator sets.

e. Accumulators

Accumulators not designed to ASME Section III are not considered a source of credible missiles since they are designed in accordance with ASME Section VIII. Various subsections of ASME Section VIII, e.g. UG-22, UG-23 and UCS-66 delineate requirements for impact testing necessary to prevent brittle fracture. The Seabrook tankage (accumulators) designed to Section VIII have either appropriate impact test, or material-operating temperatures that preclude brittle fracture. Thus, missiles from this source are not considered credible.

Various measures, such as separation of redundant safety-related equipment, have been employed to assure that essential equipment is protected against postulated missiles.

Structures, shields and barriers that are designed to withstand missile effects are tabulated in Table 3.5-1.

A tabulation of all safety-related structures, systems, and components is presented in Table 3.2-1 and Table 3.2-2. General arrangement and section detail drawings are located in Section 1.2.

3.5.1.2 Internally Generated Missiles (Inside Containment)

The principal design bases for protection against the effects of missiles generated within the reactor containment, coincident with a loss-of-coolant accident, are that:

a. Missiles generated in one reactor coolant loop shall not propagate damage to unaffected loops.

b. The function of any redundant engineered safety feature shall not be lost.

c. Containment integrity shall be maintained.

Equipment located inside containment has been evaluated for potential missile sources. As a result of this review, the following information is provided concerning potential missile sources and systems which require protection from internally generated missiles inside containment.

Catastrophic failure of the reactor vessel, steam generators, pressurizer, reactor coolant pump casings, and piping leading to the generation of missiles, is not considered credible because of material characteristics, inspections, and quality control during fabrication and erection of the particular component.

All equipment, piping and supports in containment are either secured with seismic anchors to withstand seismic disturbances or are isolated by their respective locations from safety-related equipment. Hence, there is no credible source of gravitational missiles.

The reactor coolant pump flywheel is not considered a source of missiles for the reasons discussed in Subsection 5.4.1.

Nuclear steam supply system components which are considered to have the potential for missile generation inside the reactor containment are identified as follows:

- Control rod drive mechanism housing plug, drive shaft, and the drive shaft and drive mechanism latched together
- Certain valves
- Temperature and pressure sensor assemblies
- Pressurizer heaters.

a. Control Rod Drive Mechanism Components

Gross failure of a control rod drive mechanism housing, sufficient to allow a control rod to be rapidly ejected from the core, is not considered credible for the following reasons:

- Control rod drive mechanisms are shop hydrotested at 4100 ±75 psi.
- Control rod drive mechanism housings are individually hydrotested to 3107 psi after they are installed on the reactor vessel to the head adapters, and checked again during the hydrotest of the completed Reactor Coolant System.
- Control rod drive mechanism housings are made of Type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

It is, however, postulated that the top plug on the control rod drive mechanism can become loose and be forced upward by the water jet. The following sequence of events is assumed:

1. The drive shaft and control rod cluster are forced out of the core by the differential pressure of 2500 psi across the drive shaft (the drive shaft and control rod cluster, latched together, are assumed fully inserted when the accident starts);
2. After approximately 12 feet of travel, the rod cluster control spider hits the underside of the upper support plate;
3. Upon impact, the flexure arms in the coupling joining the drive shaft and control cluster fracture, completely freeing the drive shaft from the control rod cluster;
4. The control cluster will be completely stopped by the upper support plate; however, the drive shaft can continue to be accelerated upward to hit the missile shield.

The missile shield is integrated into the simplified head assembly and consists of a reinforced two-inch thick steel plate with ventilation holes located approximately 1.5 feet above the rod travel housings. The characteristics of the postulated control rod drive missiles are given in Table 3.5-2. Penetration evaluation was performed as required by the Standard Review Plan, Section 3.5.3, Rev. 1, 1980 (NUREG 800).

b. Valves

Valves within the reactor coolant pressure boundary have been examined to identify potential missiles. It was concluded that there are no credible valve failures that could result in missile formation, since valve design features effectively preclude the ejection of valve stems. Valves are designed against bonnet-to-body connection failure and subsequent bonnet ejection by:

1. Compliance with the ASME Boiler and Pressure Vessel Code, Section III

2. Control of load during tightening of bonnet-to-body joints.

For the special case of those valves located on the top of the pressurizer that extend above the operating deck, certain missiles, although incredible, are postulated. Protection has, therefore, been provided due to potential damage to the containment liner, engineered safeguards pipes and components located outside the secondary compartments.

Missile characteristics of the bonnets of the valves in the region where the pressurizer extends above the operating deck are given in Table 3.5-3.

c. Temperature and Pressure Sensing Assemblies

The only credible source of jet-propelled missiles originating from reactor coolant piping and piping with systems connected to the Reactor Coolant System is that represented by the temperature and pressure sensor assemblies. The resistance temperature sensor assemblies can be of two types ("with well" and "without well"). Two rupture locations have been postulated: one around the welding (or thread) between the temperature element assembly and the boss for the without-well element, and the welding (or thread) between the well and the boss for the with-well element.

A temperature sensor is installed on the reactor coolant pumps close to the radial bearing assembly. A hole is drilled in the gasket and sealed on the internal end of a steel plate. In evaluating missile potential, it is assumed that this plate could break, and the pipe plug on the external end of the hole could become a missile.

In addition, it is assumed that the welding between the instrumentation well and the pressurizer wall could fail, and the well and sensor assembly could become a jet-propelled missile.

Where possible, the sensing element well is located so that if it were to become a missile, it would impact on a nearby concrete wall, floor or ceiling. The weights and impact velocities for such postulated missiles are shown in Table 3.5-4. A 10 - degree expansion half-angle water jet has been assumed. For the weights and velocities listed, they are not considered capable of generating secondary missiles upon impact with the above concrete barriers. These barriers are designed to withstand such missiles.

The missile characteristics of the piping pressure element assemblies are less severe than those of Table 3.5-4.

The missile characteristics of the reactor coolant pump temperature sensor, the instrumentation well of the pressurizer, and the pressurizer heaters are given in Table 3.5-5. A 10-degree expansion half-angle water jet has been assumed.

3.5.1.3 Turbine Missiles

a. Turbine Placement and Orientation

The placement and orientation of the turbine generators and potential missile ejection zones of ±25 degrees with respect to the low-pressure-end turbine wheels for the turbine unit is shown in Figure 3.5-1. Plan and elevation views of the Turbine Building are given in Figure 1.2-37 Figure 1.2-38, Figure 1.2-39, Figure 1.2-40, Figure 1.2-41, Figure 1.2-42, Figure 1.2-43, Figure 1.2-44 and Figure 1.2-45. Plan and layout drawings of other plant structures are also presented in Section 1.2.

b. Missile Identification and Characteristics

Experience and calculation indicate that in the improbable event of a rotor fracture, the substantial fragments of the high-pressure turbine sections and generator rotors would be contained within their respective casings (Reference 1). Accordingly, missiles analyzed in this section will be limited to postulated low-pressure turbine missiles.

1. Low-Pressure Turbine Missile Characteristics

It is considered that the failure of any of the shrunk-on wheels in the seven low pressure stages of each half of the three turbine units at both design overspeed and runaway speed can lead to the generation of external missiles.

Experience and test data indicate that the burst of a ductile disc tends to generate a small number of relatively large fragments, whereas brittle discs and wheels produce a large number of pieces with a wide variety of sizes (Reference 2). To approximate the expected number of missiles following a hypothetical nuclear turbine wheel burst, sixteen fragments in four size classes are postulated as follows: two 120-degree sectors, one 60-degree sector, three fragments weighing about one-third of the 60-degree sector, and 10 smaller pieces. It is expected that the wheels will exhibit ductile behavior and produce less than 16 fragments in the unlikely event of bursting. However, the use of the large number can be viewed as containing an allowance for smaller missiles resulting from buckets and other parts.

Postulated missile dimensions and the missile data (geometries, weights, exit energies and speed, and range of exit angles) are given in Table 3.5-6 and Table 3.5-7 (References 2 and 4).

2. Analytical Models

(a) Energy Lost in Penetrating Low Pressure Turbine Casing

The wheel is assumed to fracture into various fragments, with the fracture surfaces occurring in an axial-radial plane. The total kinetic energy of these fragments is reduced in collision with the casing structure. The methods used in determining this loss of energy are discussed in Reference 3, and are summarized below.

It is conservatively assumed that there is no loss of energy during the generating of fragments, and that there is no transfer of energy between fragments due to collision subsequent to the burst. Simple containment tests conducted by the turbine manufacturer indicate that the local shear deformation rather than gross deformation is the principal mode of energy absorption. It is assumed that the larger wheel fragments impact with and transfer energy to the stationary components of the turbine casing. An "Effective Translational Energy" is used to account for the rotational energy of the postulated missile fragments. The Stanford formula is employed in calculation of energy absorbed by the turbine casing. The wheel fragments are assumed to be oriented in the direction of minimum projected area at the instant of impact to minimize energy loss in penetration and thus, maximize the escape energy.

(b) Energy of Emerging Missiles

Exit energy and velocity ranges of emerging missiles are given in Table 3.5-7. Smaller fragments are visualized as either escaping through a hole in the turbine casing created by a larger fragment or being slowed down or stopped by the stationary turbine components. The initial size and number of fragments and the missile trajectories are dependent upon the initial burst speeds and the energy adsorption capability of the stationary components. The assumptions used in obtaining the missile exit parameters result in conservative estimates of these parameters.

(c) Missile Ejection Angles

Missiles generated by interior stages are ejected within a narrow angular zone (-5.0° to 5.0°) with respect to the turbine axis. End stages, on the other hand, are ejected into angles between 0.0° and 25.0° with respect to the turbine axis, away from the adjacent intermediate stages. In addition, the low pressure turbines rest upon massive reinforced concrete supports, 10 ft thick. These supports will effectively screen out all missiles emerging at angles below -14.5° (with respect to the horizontal).

c. Target Description

The characteristics of the safety-related areas subject to unacceptable damage by turbine missiles are provided in Table 3.5-8. Discussions of specific treatments are provided below. Note that no safety-related structure in the plant lies within the -5° to 5° direct missile zone for missiles from intermediate stages.

1. Containment

The containment is modeled as a flat-topped cylinder having the same horizontal and vertical projected area as the domed structure. The presence of the secondary containment is ignored in the assessment of damage probabilities, since it will have little effect upon the very energetic missiles postulated above.

Those turbine missiles identified as possessing sufficient energy to perforate the containment shell have been evaluated as part of a probabilistic study which has determined that these missiles fall into an established category of acceptable risk (see Updated FSAR Subsection 3.5.1.3d) and, as such, are not a design consideration for the containment structure. The remaining turbine missiles do not possess sufficient energy to perforate the containment shell. Some may possess sufficient energy to cause dislodgement of concrete on the inside face, local to the impact area, but the liner plate will serve to contain these concrete fragments, thus preventing any secondary missiles from entering the containment.

2. Control Building

The damage criterion for the Control Building is taken to be barrier spall. No credit is taken for redundancy of vital systems in the 21.5 ft and 50.0 ft levels, since these redundant trains are not separated by missile barriers.

3. Diesel Generator Building

The diesel generators are located at the 21.5 ft - level and are separated by a 2 - foot thick missile barrier. It is not possible for any direct missile to penetrate an external wall protecting one train and continue on to damage directly, or by secondary missiles, the second train. At the 51.5 ft - level, the redundant trains are not separated by missile barriers so that the damage criterion is taken to be small.

4. Condensate Storage Tank

The condensate storage tank consists of an inner stainless steel tank surrounded by a reinforced concrete barrier which acts as an outer tank. This dual tank arrangement is required to contain a minimum of 200,000 gallons of water. The elevation below which perforation of the outer tank is unacceptable is 47.5 feet (which includes margin to compensate for missile effects). It is conservatively assumed that perforation of the outer concrete tank will generate spall fragments which will perforate the steel tank.

5. Other Buildings

The damage criterion for other buildings (Emergency Feedwater Pump Building, Equipment Vaults, Service Water Pumphouse, Primary Auxiliary Building) is taken to be spall of the external missile barriers.

d. Turbine Missile Strike and Damage Probability Analysis

Turbine missiles strike probability (P_2) is calculated using the UE&C computer code "TURB1," which is based upon the methods developed in Reference 5. For a given set of missile ejection conditions, the fraction of permissible trajectories striking any portion of a safety-related building is determined. Suitable averages over the ejection conditions are taken to obtain the estimates of P_2.

The probability of target damage, given a missile strike, P_3, is obtained by determining the fraction of missile trajectories striking a barrier which can result in damage with radiological consequences exceeding regulatory. In most cases, the damage criterion used is barrier spall and the stopping power of intermediate barriers is determined by Modified NDRC Missile Damage Equations (Reference 6).

1. <u>Analytical Model</u>

It is assumed that one, and only one, turbine stage will fail under any set of overspeed conditions. The basis for this assumption is that the failure of the stage in question will result in an immediate end to the overspeed condition responsible for the forces resulting in stage failure.

Also, since missiles are directed perpendicular to the turbine axes, it is unlikely that a missile from a failed stage will strike an adjacent stage, thus possibly resulting in cascading failures (Reference 1).

The basic equation for the analysis of turbine missiles is:

$$P_4 = P_1 P_2 P_3 \tag{1}$$

where P_1 is the rate of turbine failure (events per year) resulting in the generation and ejection of missiles, P_2 is the conditional probability that a missile will strike a specified target, given its generation and ejection, and P_3 is the conditional probability that the missile will cause damage to the target that may lead to unacceptable consequences, given that the target is hit. P_4 is then the probability (events per year) of occurrence of unacceptable damage from a turbine failure.

Two estimates of P_1, provided by the turbine manufacturer (Reference 1), are available, as shown below:

P_1 (Normal overspeed) Low Trajectory = $1.4\text{x}10^{-6}$/year

P_1 (Normal overspeed) High Trajectory = $4.0\text{x}10^{-5}$/year (2)

P_1 (Runaway overspeed) Low Trajectory = $1.4\text{x}10^{-6}$/year

P_1 (Runaway overspeed) High Trajectory = $4.0\text{x}10^{-5}$/year

The estimates provided by the turbine manufacturer are based upon a detailed evaluation of the probabilities of the failure mechanisms which may lead to the various overspeed conditions. The entire failure probability is conservatively applied to each estimate. The analysis uses stage 7A only for low trajectories and all stages for high trajectories.

The P_l estimates provided by the turbine manufacturer are based, in part, on the testing and maintenance frequencies of the turbine stop, control, and combined intercept valves. MPR Report 0326-0097-RPT-001, Seabrook Turbine Valve Maintenance and Test Interval Assessment, provides an evaluation of the valve unreliabilities as a function of the testing and maintenance intervals. Based on this assessment, new P_l estimates have been developed (Ref, Engineering Evaluation EE 19-001) based upon an extended control valve testing interval of 5 months vs the 3 month interval recommended by the turbine manufacturer. These P_l estimates and the resultant damage probabilities are reflected in Table 3.5-9 and 3.5-10.

The specific criteria for acceptability of the level of risk from turbine-generated missiles may be stated as:

(a) High Trajectory Missiles

The probability of a high trajectory missile strike on any safety-related area (i.e., any structure in which missile induced damage may lead to radiological consequences in excess of regulatory limits) shall be less than about 10^{-7} per year per area.

(b) Low Trajectory Missiles

The probability that a low trajectory missile strike results in radiological consequences exceeding regulatory limits should be less than about 10^{-6} per year per unit for all safety-related areas in that particular unit. This criterion is contingent upon the demonstration that the method of calculation of probabilities is conservative.

In the present analysis, the probability estimates are based upon SRP 3.5.1.3 turbine failure probabilities. In view of the Turbine Overspeed Protection Systems described in Section 10.2, and because of the extensive quality assurance program of the manufacturer, these failure rates are considered to be most conservative.

Additional conservatism is due to the assumption that barrier damage will generally lead to radiological consequences exceeding regulatory limits. No credit is taken for redundancy, unless redundant trains are separated by missile barriers which would contain spall fragments.

It is assumed, that the failure of a given stage will result in the emission of a number N_L of k types of missiles characterized by a weight, W_i. The probability that the i-th type of missile will strike a given target depends upon the energy, E, and angles of emission. The probability distribution for energy, P_E, is taken to be uniform.

$$P(E_i)dE_i=dE_i/(E_{2i}-E_{2i}),E_{1i}\leq E_i\leq E_{2i} \quad (3)$$

$$= 0 \qquad ,\text{otherwise}$$

The probability that a missile will be ejected into a solid angle element, defined by the polar angle θ, that (measured from the perpendicular to the turbine axis) and the azimuthal angle ∝ (measured around the polar axis) is also taken to be uniform (see Figure 3.5-2).

$$P'_{\Omega}d\Omega'=\cos\theta d\theta d\alpha/[2\pi(\sin\delta_U-\sin\delta_L)],\delta_L\leq\theta\leq\delta_U \quad (4)$$

$$= 0 \qquad ,\text{otherwise}$$

The upper and lower angle constraints are dependent upon the specific emission conditions. Table 3.5-7 contains values of N_L,W_i,E_{li},E_{2i},k,δ_U, and δ_L for each stage type for the two sets of overspeed conditions (120% and 180%).

It is necessary to relate the solid angle of emission, dΩ', to the solid angle of emission, dΩ, in the coordinate system in which the azimuthal angle, ψ, is measured horizontally from the turbine axis and the polar angle, ø, is the angle between the initial velocity vector and the horizontal plane. (See Figure 3.5-2 for angle definitions.) Thus,

$$d\Omega'(\alpha,\theta)=J()d\Omega(\psi,ø) \quad (5)$$

where J is the Jacobian of the transformation. It is found that:

$$\cos\theta d\theta d\alpha = \cos ø dø d\psi \quad (6)$$

where:

$$\cos\theta = \cos ø \cos\psi \quad (7)$$

By combining equations (4), (5), (6) and (7), it is found that:

$$P_{\Omega} d\Omega = \frac{\cos\varphi os\varphi d}{2\pi\left(\sin\delta_u - \sin\delta_L\right)}, \sin\delta_L \le \cos\varphi\cos\psi \le \sin\delta_U \le \text{ and } \varphi \ge -14.5° \quad (8)$$

The additional condition in Equation (8) arises from the barrier present at the turbine. It is now necessary to relate the target coordinates to the emission angle ø. Assume that a target area increment is located at polar coordinates (ψ, R, Y) and the turbine stage at coordinates (O, O, Y_T). Also,

$$V_i^2/g = 2\ E_i/W_I \quad (9)$$

where V_i is the missile ejection velocity and g the acceleration of gravity. The trajectory angle, ø, required to hit the target element is:

$$\tan\varphi = \frac{2E_i}{W_i R} \pm \left[\left(\frac{2E_i}{W_i R}\right)^2 - 1 - \frac{4E_i}{W_i}\ \frac{(Y - YT)}{R^2}\right]^{1/4} \quad (10)$$

The positive sign in Equation (10) reference to "high trajectory" missiles, the negative sign to "low trajectory" missiles. The constraints given in Equation (6) may represent missile strikes of either type, dependent upon the location of the area element. Equation (10) also implies a range constraint in that the interior of the square root must be nonnegative. A ground-level target area whose coordinates are such that a low trajectory missile may strike the target is said to be in the direct zone of that missile, or, since the emission angle constraints are the same for all missiles ejected by a given stage, the target is in the direct zone of the stage. Targets in the direct zone of any turbine stage are considered to be in the direct zone of the turbine unit, as shown in Figure 3.5-1. Targets outside the direct zone of the turbine may be struck by low trajectory missiles under certain conditions. Such strikes are considered direct strikes for the purposes of this program. In the following, let J equal conditions of the missile strike, and consist of the following levels:

(a) Turbine number

(b) Building number

(c) Type of trajectory (low, high)

(d) Type of hit (direct, indirect)

(e) Portion of building struck (wall, roof).

Let N be the number of stages (42 for 180 percent overspeed, 6 for 120 percent overspeed in the present case). The probability of a strike P_2 (J) may be written,

$$P_2(J) = \frac{1}{N_S} \sum_{1=1}^{N_S} \sum_{1=1}^{k} N_k \int_{E_{2i}-E_{li}} dE_i \int_{Aj,i,l(E_i, Xl)} \frac{P\Omega}{dA} \frac{d\Omega}{dA}\, dA \qquad (11)$$

where the interior integral is performed over all building area increments such that a hit of type (c), (d) and (e) is permissible, given the constraints of Equations (8) and (10). These constraints are to be applied at the coordinates x_i representing the position of the area increment with respect to the postulated failed disc 1.

An expression similar to Equation (11) may be derived for

P_2 (J) $\bullet$ P_3 (J) by changing P_Ω to P'_Ω where:

$$P'\Omega = Q\Omega P_\Omega \tag{12}$$

P_3 (J) is the probability of damage, given a missile strike, and Q_Ω is the conditional probability that a missile whose parameters permit a strike on the target increment will result in unacceptable damage, as determined by the modified NDRC Equations shown below:

$$G(X/d) = kNd^{0.2}(w_i/d^3)(V_i'/1000)^{1.8} \tag{13}$$

where V'_i is the component of velocity normal to the surface of impact (ft/sec), d is the mean effective diameter of the missile in inches, N = .84 for blunt missiles, and

$$k = 180/\sqrt{f_c'} \tag{14}$$

where f_c' is the concrete strength in psi. Then:

$$\begin{aligned} X/d &= 2\sqrt{G(X/d)}, \qquad G(X/d) \le 1 \\ &= 1 + G(X/d), \qquad G(X/d) \le 1 \end{aligned} \tag{15}$$

In the above, x is the penetration distance for the missile. Normally, d is taken to be the diameter of the circle whose area is the mean area of the missile fragment. For the missiles of interest in this program, the target damage depends only slightly upon the value of d used.

Equations (13) - (15) are used to determine the effectiveness of intermediate barriers in stopping missiles by determining the amount of energy lost in the barrier. For the purposes of this analysis, only structures specifically designed as missile barriers are postulated to have an effect upon missile trajectories. Most intermediate vertical barriers are most likely to be struck by missiles with high angle of incidence (i.e., greater than 65° with respect to the surface normal).

Although such missiles may result in unacceptable damage to those systems protected by the intermediate barrier, it is most unlikely that these missiles will cause damage after a second impact.

Additional clarification of this point is provided in Subsection 3.5.1.3c.

In several cases, the specific criterion used to determine acceptability of damage is that the missile barrier shall not be perforated. The modified NDRC Barrier Perforation Equations are:

$$\begin{aligned} t_p/d &= 3.19\,X/d - 0.718(X/d)^2, \; X/d \le 1.35 \\ &= 1.32 + 1.24\;X/d \qquad\quad , X/d \le 1.35 \end{aligned} \tag{16}$$

The barrier thickness required to prevent perforation is t_p. The spall distance (the thickness of concrete required to prevent spall for missiles of a given type) is given by:

$$\begin{aligned} t_s/d &= 7.91\,X/d - 5.06(X/d)^2, \; X/d \quad 0.65 \\ &= 2.12 + 1.36\;(X/d) \qquad , X/d \quad 0.65 \end{aligned} \tag{17}$$

The damage probability Q_Ω is assumed to be:

$$\begin{aligned} Q_\Omega &= 1, \qquad tp, s \ge t \\ &= 0, \qquad tp, s < t \end{aligned} \tag{18}$$

where t is the thickness of the barrier at the missile impact point. Equations (13), (16) and (17) are applied on an area by area basis.

2. <u>Application of the Model</u>

The computer program TURB1 consists of a set of nested loops designed to perform the integrals, averages, summations and damage and strike categorizations implied in Equation (11). Each stage/building combination is divided into the area increments by angle $\Delta\psi$ (approximately one degree in extent) and vertical or radial dimensions (five increments in either case). Thus, the containment for Unit 1 is divided into approximately 34 angle increments. The initial missile energy range is modified to eliminate those portions for which the angle constraints preclude a hit on the target area in question. This modified range is then divided into five increments. Each combination of energy and area elements is then tested to determine which types of hits (and damage) are permissible according to the angle constraints.

Sensitivity analyses conducted on the angle, lateral dimension and velocity incrementing, indicate that this increment size is sufficient to reduce computational error to a few percent.

3. <u>Missile Shielding</u>

In this analysis, it is assumed that only reinforced concrete structures specifically designed as missile barriers can have any effect upon missile trajectories. In many cases, a safety-related structure having such a barrier lies on the direct line between the turbine and another safety-related building. Any low trajectory missile striking the second building will necessarily result in unacceptable damage to the first building. To avoid double-counting of unacceptable missile strikes, it is necessary to take such shielding into account.

The condensate storage tank is a special case, since impacts on its surface above 47.5 feet will not cause unacceptable damage. In this case, it is assumed that missiles can penetrate the condensate storage tank (losing energy in the process) and continue on to possibly damage other safety-related buildings. The total energy lost is taken to be the energy required to penetrate two concrete barriers (as calculated by the Modified NDRC Penetration Equations) each two feet thick. No credit is taken for oblique impacts or the stopping power of the inner tank.

In some cases, it is possible that a missile will strike a building wall at an oblique angle and ricochet without causing any damage to the building. The building arrangement ensures that it is most unlikely that such deflections can impact on another safety-related structure. This point ensures conservatism in the analysis.

4. <u>Results</u>

The results of the analysis are shown in Table 3.5-9, (low trajectory missiles) and Table 3.5-10 (high trajectory missiles). The total probability of unacceptable damage to safety-related structures from low trajectory missiles is less than $1x10^{-7}$ per year. The probability of a high-trajectory missile hit is less than $1x10^{-7}$ per year for any single structure. Many such missile strikes will not result in unacceptable damage.

It is concluded, therefore, that the plant layout and design provides adequate protection from turbine missiles.

e. Turbine Overspeed Protection

A description of the Turbine Overspeed Protection System in terms of redundancy, diversity, component reliability, and testing procedures, is provided in Subsection 10.2.2.

f. Turbine Valve Testing

A discussion of the turbine valve testing procedures and test frequency is provided in Section 10.2.

g. Turbine Characteristics

Turbine data pertinent to the evaluation of its failure characteristics, including a description of its overall configuration, major components (e.g., steam valves, reheaters, etc.), rotor materials and their properties, steam environment (e.g., pressure, temperature, quality, chemistry), and other appropriate properties are provided in Section 10.2. Turbine operational and transient characteristics, including turbine startup and trip environment, as well as its overspeed parameters are also provided in Section 10.2.

3.5.1.4 Missiles Generated by Natural Phenomena

Possible design basis tornado-generated missiles that could originate at the site are listed in Table 3.5-11 along with their dimensions, weights and associated impact velocities. This missile spectrum is taken from Standard Review Plan 3.5.1.4, Rev. 1, Missiles Generated by Natural Phenomena (6/77), paragraph 5, designated as "SRP 3.5.1.4, Rev. 0, Missile Spectrum." (This spectrum is identical to that defined as "SRP Subsection 3.5.1.4, November 24, 1975 Missile Spectrum A" in NUREG-0800, SRP 3.5.1.4, Rev. 2, July 1981.)

These missiles are considered to be capable of striking in all directions with vertical velocities equal to 80 percent of the acceptable horizontal velocities. Missiles A, B, C, D and E are to be considered at all elevations, and missiles F and G at elevations up to 30 feet above all grade levels within one-half mile of the facility structures.

The structures and/or barriers designed to resist tornado-generated missiles are tabulated in Table 3.5-12. Information on the protected systems or components is included in Subsection 3.5.2. All walls and slabs used for missile protection are constructed of reinforced concrete having a minimum thickness of two feet and a minimum specified 28-day compression strength of 3000 PSI.

The containment enclosure structure, which is constructed of reinforced concrete with a minimum thickness of 15 inches, will not be breached by tornado-generated missiles.

The design procedures discussed above will ensure that the probability of a tornado missile impact on a missile barrier leading to exceeding 10 CFR 100 limitations will be less than 10^{-7} per year. In addition, evaluations (References 20 and 29) of a missile entering a safety-related structure resulted in a probability of about 10^{-6} per year.

3.5.1.5 Missiles Generated by Events near the Site

As stated in Subsection 2.2.3.1, missiles generated by explosions near the site are extremely unlikely to reach the site area. Therefore, no missile impacts from these sources need be postulated.

3.5.1.6 Aircraft Hazards

a. The Model

As discussed in Subsection 2.2.2.5, it has been determined that the method used by the NRC (Reference 8) in its evaluation of the Boardman Nuclear Plant is applicable to the Seabrook site.

The model treats each aircraft operating from an airport or on an airway individually, finds the probability of impact for each plane and sums up all the individual probabilities to obtain the total crash probability:

$$P = \sum_{i=1}^{K} N_i A_i R_i \left[\frac{\text{Exposure}}{\text{Random Transit}} \right]_i \tag{1}$$

where i = Summing index for K different aircraft

N_i = Number of flights of aircraft "i" per year that have a finite probability of impacting the site

A_i = Area of critical structures at the site (mi^2)

R_i = Accident rate of aircraft "i" (acc/hr)

$\left[\frac{\text{Exposure}}{\text{Random Transit}} \right]_i$ = Time per square mile that aircraft "i" is in the vicinity of the site (hr/mi^2)

b. Area of Critical Structures (A)

The area is determined by assuming a thirty degree slope for a disabled aircraft.

The structures considered as a target include:

1. Containment Building
2. Fuel Storage Building
3. Primary Auxiliary Building
4. Control Building
5. Diesel Generator Building
6. Refueling water storage tank
7. Reactor makeup water storage tank

The Turbine, Heater Bay, Waste Processing and Administration Buildings are not considered to be targets and are assumed to serve as a shield for critical buildings behind them. Figure 3.5-3 is a sketch of the plot plan and shows the effective target area. The target area shown on this figure was reduced by eliminating the Service Water Pumphouse which is backed up by the ultimate heat sink cooling tower 1000 feet away. Either one of these two structures can supply cooling water. Therefore, because of this redundancy and separation, they were not considered in the target area. The resultant target area is 0.005 square miles per unit.

The Containment Building is considered as part of the target area for aircraft weighing more than 81,800 pounds. Based on a structural analysis, the containment can withstand the impact of an FB-111A and smaller aircraft. Therefore, the target area has been reduced for these aircraft to an area of 0.0029 square miles to include the other critical structures only. The structural analysis, presented in Appendix 2P, verifies the containment integrity after an 81,800 pound aircraft impact of the FB-111A type. It was also shown by analysis that all the critical structures can withstand the impact of small aircraft equal to or less than 12,500 pounds.

c. Exposure Per Random Transit

The exposure per random transit is $\frac{1}{2vgh}$, where "v" is the speed of the aircraft while in the vicinity of the site, "g" is the glide ratio of the particular aircraft and "h" is the altitude of the aircraft. A glide ratio of 17 has been assumed for all military and commercial aircraft excepting the FB-111A. A glide ratio of 13 is assumed for the FB-111A.

d. Federal Airways

A description of the federal airways and a conservative estimate of the number of aircraft that pass within 2 miles of Seabrook can be found in Subsection 2.2.2.5. FAA area specialists (Reference 9) examined routing slips of each aircraft that flew in the Pease Sector on the peak traffic day of 1979 (July 27, 1979). They determined that 224 flights could have actually passed within 2 miles of the site. Of the 224 flights, 128 aircraft weighed more than 12,500 pounds. A cruise airspeed for each aircraft was determined (Reference 10), and from this an average airspeed was computed. The average altitude for the 128 aircraft was 30,000 feet. Based upon 128 possible overflights of aircraft over 12,500 pounds on the peak traffic day, an annual estimate of 46,720 overflights can be computed. For commercial aircraft:

$$\text{Exp/Trans} = \frac{1}{2vgh}$$

$$= \frac{1}{2(500)(17) \quad (5.6818)}$$

$$= 1.04 \times 10^{-5}\ \text{hr}/\text{mi}^2$$

e. Airports Within 5 Miles

Hampton Airport is described in Subsection 2.2.2.5. As stated, the aircraft at Hampton are single engine and therefore weigh considerably less than 12,500 pounds. It can be assumed that since the airport consists of a very short turf runway, it is highly unlikely that an aircraft weighing more than 12,500 pounds would use this airport. Since the critical structures at Seabrook can withstand the impact of aircraft weighing less than 12,500 pounds, the risk from this source is zero.

f. Airports Beyond 5 Miles

Subsection 2.2.2.5 describes airports beyond 5 miles and their associated risk to the Seabrook site.

g. Military Installations

A description of the traffic patterns, aircraft, accident rates, and annual number of transits proximal to Seabrook for each type aircraft using Pease Air Force Base can be found in Subsection 2.2.2.5.

The traffic pattern altitude which brings aircraft operating out of PAFB in proximity to Seabrook is 2,000 feet. A downwind airspeed of 220 miles per hour is assumed for the FB-111A, and 200 miles per hour for all other military aircraft.

For FB-111A aircraft:

$$\text{Exp/Trans} = \frac{1}{2_{vgh}}$$

$$= \frac{1}{2\,(220)\,(13)\,\,(.379)}$$

$$= 4.61\times10^{-4}\,\text{hr}/\text{mi}^2$$

For all other military aircraft:

$$\text{Exp/Trans} = \frac{1}{2\,(200)\,(17)\,(.379)}$$

$$3.89\times10^{-4}\,\text{hr}/\text{mi}^2$$

h. Accident Probability

Using equation (1), the total accident probability is calculated to be $9.71\text{x}10^{-8}$. Table 3.5-13 shows the inputs to the calculation of this result. The number of operations and accident rates for each type of aircraft can be found in Subsection 2.2.2.5.

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

All plant structures, systems and components whose failure could lead to offsite radiological consequences, or which are required to shut down the reactor and maintain it in a safe condition while assuming an additional single failure, are designated ANS Safety Class 1, 2 or 3 and/or seismic Category I, and are listed in Table 3.2-1 and Table 3.2-2.

Safety-related components, including essential piping, instrumentation and electrical equipment, are protected against damage from externally generated missiles by physical barriers or protective structures.

The structures protecting the systems and components important to safety are listed in Table 3.5-12; plan and elevation drawings pertinent to these structures are found in Section 1.2.

Discussions on design requirements which exempt the refueling water storage tank and spray additive tank from missile protection are presented in Subsection 6.2.2.

Protection of the Fuel Storage Building spent fuel pool cooling/cleanup system and storage pool from missiles which could penetrate the nonmissile-proof fuel shipping cask rail car access door is provided by an interior missile-proof wall, as shown in Figure 1.2-16. Results of the probabilistic analysis (Reference 20) indicate that the overall entrance-probability for the Fuel Storage Building is less than $3x10^{-9}$ per year.

The ultimate heat sink complex (Subsection 9.2.5), which consists of the mechanical draft cooling towers and the Atlantic Ocean service water system, has appropriate portions protected against all credible missiles. Those portions of the intake and discharge transition structures which house the safety-related service water valves (valve pit area) are designed to withstand tornado generated missiles. Since the safety-related portions of the piping entering the transition structures are enclosed and protected as they pass through this area, only the pipe entrance to the structure is exposed. However, the pipes are located under water at or below elevation (-)35'-0". The pipes are, therefore, not exposed to credible tornado-generated missiles.

Referring to Standard Review Plan 3.5.1.4, "Missiles Generated by Natural Phenomena," the largest missiles which might be considered to potentially block the pipe entrance are the 35-foot long utility pole and the 4000 lb. automobile. In order to block the pipe entrance, the automobile would have to be directed through the top of the transition structure, missing the concrete cross-bracing, and drop exactly at the pipe location, sinking to the elevation of the pipe. Since the pipe is located at elevation (-)40'-0" (in the intake transition structure), and the floor is at elevation (-)55'-0" the automobile would continue to sink to the floor where it could not pose any blockage to the pipe. The utility pole would have to be similarly directed, however, it would pose no blockage problem since it would float at water level considerably above the elevation of the pipe.

The only safety feature of the cooling tower is to function in the event of a seismic disturbance of sufficient magnitude to interrupt the flow in the main circulating water tunnels. Accordingly, only those parts of the Service Water Cooling Tower structure which protect service water piping up to and including the cooling tower pump discharge valves require missile protection. Certain parts of the tower and its structure are not protected against missiles. These include the tower air intakes, tower pump, fans and gear boxes and electrical switchgear. The remainder of the service water equipment and piping is located within the missile-protected Service Water Pumphouse, the Primary Auxiliary Building, or buried underground. The sink safety function is assured through the use of the missile-protected Service Water System and its seismic Category I cooling tower which provides an alternate source of cooling water. Refer to Subsection 9.2.5 for further details concerning the ultimate heat sink.

3.5.3 Barrier Design Procedures

The structures and barriers identified in Subsection 3.5.2 are designed to withstand the local effects and overall effects of the applicable missiles. The following areas are discussed:

a. Methods for the prediction of local damage in the impacted area, including estimation of the depth of penetration and, in case of concrete barriers, the potential for generation of secondary missiles by spalling or scabbing effects.

b. Methods for the prediction of the overall response of the structure or barrier due to the missile impact. This includes assumptions on acceptable ductility ratios where elasto-plastic behavior is relied upon, and procedures for estimation of forces, moments, and shears induced in the barrier by the impact force of the missile.

3.5.3.1 Local Damage Prediction

a. In Concrete

The modified Petry equation, as given by A. Amirikian (Reference 11), has been commonly used to estimate missile penetration into concrete barriers. Sufficient thickness of concrete is required to prevent perforation and scabbing.

Modified Petry's formula for penetration is:

$$D = KA_pV'$$

where:

D = penetration in a barrier of thickness, t, where $t \geq 3D$, both t and D in feet

K = coefficient for penetration for the material

= 3.50×10^{-3} for reinforced concrete ($f'_c = 3000$ psi)

A_p = sectional pressure

= W/A lbs./sq ft

W = weight of missile in lbs.

A = cross section of missile in ft^2

V' = $\log_{10}\left(1 + \frac{V^2}{215,000}\right)$

V = velocity of missile in ft/sec

For barriers having thickness $t < 3D$, the penetration D' was calculated as:

$$D' = D\left[1 + e^{-4(a'-2)}\right]$$

where:

$a' = t/D$

The minimum thickness required to prevent scabbing for steel missiles having solid circular cross section was determined from Figure 3.5-4.

For other noncompressible missiles, design for no scabbing was based on $t \geq 3D$, Reference 11.

Since the PSAR, experimental data has been generated by Sandia Laboratory for the Electric Power Research Institute for the specific missiles, velocities and targets applicable to these structures and barriers. The semi-analytical, semi-empirical modified National Defense Research Committee (NDRC) equations have been found to be more accurate than the modified Petry equations, Reference 12.

The Modified NDRC formula for penetration is:

$$x = \sqrt{4\ K\ N\ W\ D\left(\frac{V}{1000d}\right)1.8} \qquad \text{for } x/d \leq 2.0$$

where:

x = penetration depth in inches

K = $180/\sqrt{f'_c}$ where f'_c is the concrete design strength

N = missile shape factor

= 0.72 for a flat nose missile

W = weight of missile in lbs.

d = diameter of missile in inches

V = striking velocity of missile in ft/sec

The minimum thickness to prevent scabbing is given by:

$$t_s/d = 2.12 + 1.36\ x/d \text{ for } 0.65 \leq x/d \leq 11.75$$

where for pipes:

$d = d_e$ = effective diameter as though a solid cylinder

A factor of safety of 1.2 was applied to t_s.

The design thickness of walls and roof slabs was checked for its adequacy to prevent scabbing using both sets of equations and was found to be satisfactory.

b. In Steel

Tornado doors are 2" thick A-36 steel barriers designed to resist tornado-generated missiles in addition to tornado wind pressures (Reference 21). Maximum penetration depths are calculated by the BRC formula (Reference 15) to be 0.84 inch for the 6" diameter steel pipe and ~0.80 inch for both the 12" diameter steel pipe and 1" diameter steel rod.

3.5.3.2 Overall Damage Prediction

Concrete barriers and structures subject to missile impact were designed to sustain the dynamic force of impact without shear or flexural failures, or excessive deformation. The dynamic force is a function of many parameters including the mass, velocity and rigidity of the missile, the natural period of vibration, stiffness and ductility of the target. The missiles can be conveniently grouped into two basic categories, rigid (nondeforming) and nonrigid (deforming). Examples of each group include turbine parts and tornado-borne pipes and rebar for the first, and aircraft and tornado-borne wood poles and automobiles for the second. A structure or barrier impacted by a missile of either group was analyzed for its maximum response to a force-time loading function using the methods in References 13, 14 and 15. The problem is idealized as an equivalent one-degree-of-freedom elasto-plastic structure subjected to a suddenly applied load of given pulse shape, duration and magnitude. The force-time load histories for the rigid pipe and rebar missiles are calculated using a modified Williamson and Alvey method (References 15 and 16), while those for the wood pole and aircraft are calculated per Riera (Reference 17), and that for the auto per Bechtel's BC-TOP-9A (Reference 18).

The methodology used for the design of the exterior missile shield at the South East corner entranceway to the Service Water Pump House is described in subsection 3.5.3.2.d.

a. Missile Force-Time Histories

1. Pipes, Rebar and Other Rigid Missiles that Penetrate

The missile is assumed to penetrate the barrier with linearly decreasing velocity and constant force. The resulting rectangular pulse forcing function has a magnitude, F, and duration, t_d, calculated as:

$$F = \frac{1}{2}\frac{w}{g}\frac{V_o^{\ 2}}{x}$$

and

$$t_d = \frac{2_x}{V_o}$$

where:

W = weight of missile (lbs)

V_O = impact velocity (ips)

g = 386.4 in/sec^2

x = penetration depth (in)

The penetration depth was calculated by the modified NDRC equation which has been shown to give the best correlation to experimental data.

The pulse magnitudes and durations for the 743 lb., 12" steel pipe are as follows:

Direction	Velocity (fps)	F (Kips)	t_d (sec.)
Horizontal	211	743	0.00656
Vertical	169	582	0.00671

2. Utility Pole

A force-time history was developed for the utility pole using the Riera formula, Reference 17 as follows:

$$F(t) = F_c A + \mu V^2(t)$$

where:

F_c = the "crushing strength" of the wood pole (lb./ft^2)

A = the cross-sectional area (ft^2)

μ = the mass per unit length (slug/ft)

V(t) = the instantaneous velocity of the missile during impact (ft/sec)

Experimental data from the Sandia tests for EPRI, Reference 19, were used to define a reasonable value of crushing strength. The above equation was integrated in time for the definition of F(t). The forcing function was then approximated by a rectangular pulse as:

Direction	Velocity (fps)	F (Kips)	t_d (sec.)
Horizontal	211	202.5	0.054
Vertical	169	192	0.044

3. Auto

The force-time history defined in Bechtel's report BC-TOP-9A, Reference 18, was used for the auto impact.

$$F(t) = 0.625\, V_s\, W_m \sin 20t \qquad 0 \le t \le 0.0785 \text{ sec}$$

$$= 0 \qquad t > 0.0785 \text{ sec}$$

where:

V_s = the initial impact velocity (ft/sec)

W_m = weight of auto (lbs.)

This is a quarter sine wave with peak force of 265 Kips and duration of 0.0785 seconds for V_s equal to 106 ft/sec.

4. Aircraft

For a discussion on aircraft hazards and the ability of seismic Category I structures, particularly the containment, to withstand postulated impacts from aircraft, see Subsection 3.5.1.6, Aircraft Hazards.

b. Procedure for Calculating Overall Response

The procedures used in calculating structural response of walls and slabs to the impact forces of the rigid missiles, utility pole and auto are those described in References 13, 14 and 15, summarized as follows:

1. The load pulse shape, peak force, F, and duration, t_d, are defined as above.

2. The slab (or beam) capacity, R_m, natural period of vibration, T, and maximum allowable ductility, μ, are calculated or defined.

3. The effect of other forces acting simultaneously with the impact force on the structure or barrier are included. The displacement, X_s, of the structure under these static loads and an effective point force, R_s, are calculated.

4. The capacity of the structure to resist the impactive force is defined as $R'_m = R_m - R_s$ and an effective allowable ductility is defined as $\mu' = (X_m - X_s)/(X_e - X_s)$ where X_e is the yield displacement and $X_m = \mu X_e$.

5. The system behavior is assumed elasto-plastic with an equivalent single degree-of-freedom. Ratios $C_T = t_d/T$ and $C_R = R'_m/F$ are formed. Figure 3.5-5, a chart of X_m/X_e curves for rectangular impulse loads, or similar figures for the appropriate pulse shape are entered with the values of C_T and C_R, and the displacement response measure, X_m/X_e, is determined.

   This measure is compared to μ as follows:

   X_m/X_e $\leq\mu$: no failure, acceptable

   $>\mu$: unacceptable

   Additionally

   X_m/X_e ≤ 1 : elastic response

   >1 : permanent displacement

The relative structural response to the tornado-borne rigid missiles, pipes and rebar, were compared. The 12"ϕ pipe produced the greatest deformations in all cases. Hence, subsequent analysis of overall structural effects disregarded the smaller pipes and rebar.

c. Permissible Ductility Ratios

Reference 14 was followed for definition of permissible ductility ratios of reinforced concrete members.

1. For beams, walls or slabs where flexure controls design:

μ = $0.05/(\rho - \rho') \leq 10$

where:

ρ = A_s/bd, $\rho' = A_s'/bd$

A_s = area of tension reinforcement

A_s' = area of compression reinforcement

d = distance from extreme compression fiber to centroid of tension reinforcing steel

b = width of beam or unit width of slab

For flexure to control design, the load capacity of a structural element in shear shall be at least 20 percent greater than the load capacity in flexure. Punching shear capacity of reinforced slabs under dynamic loading has been shown to be greater than that under static loading. The ultimate shear stress was assumed as $V_u = 4.4 \sqrt{f'_c}$ Rigid missiles (e.g., pipes and rebar) which are shown not to perforate a barrier or structure shall be considered as pro-ducing flexural controlled response in the structure.

2. For beams, walls and slabs where shear controls design and shear is carried by concrete alone, $\mu = 1.3$.

3. For axial compression, $\mu = 1.3$.

The following permissible ductility ratios were used for steel members:

1. For flexural members, $\mu = 0.2 \ \in_u/\in_y$

   where:

   $\in_u$ = ultimate strain at rupture

   $\in_y$ = yield strain

2. For tension members, $\mu = 0.5 \ \in_u/\in_y$

3. For compression members, $\mu = 1.0$

d. Structural Capacity and Period of Vibration

The ultimate moment capacity of walls, slabs and beams is calculated using provisions of the appropriate codes. The collapse load for a point load was related to the ultimate moments by yield-line theory as in Reference 15.

The natural period of vibration of the "equivalent" single degree-of-freedom structure was calculated to correspond to that of the first flexural mode of the actual structure. The effective moments of inertia of concrete beams, walls and slabs were assumed to be the average of the gross and cracked moments of inertia.

The Service Water Pump House (SWPH) is used to store required FLEX portable equipment as part of the station’s response and commitments made to NRC Order EA-12-049 (Ref. 22) involving the Fukushima event. To facilitate FLEX equipment storage within the SWPH, the existing building entranceway at grade was modified to permit straight in-out FLEX equipment access. Due to this change, a tornado missile barrier was added to protect safety related components inside the SWPH (refer to UFSAR Section 3.5.2 for a description of the SSCs requiring protection) as well as the stored FLEX equipment.

The tornado missile barrier is an exterior structure that is stand-alone and not physically attached to the SWPH (Ref. 24). It is primarily a steel framed structure, with primary load carrying members fabricated from steel plate, cable, fabric strap and structural tube members. The vertical post portions of the barrier structure are embedded in a reinforced concrete foundation. The barrier has a manually operated gate to permit equipment access to the SWPH. The barrier is designed for tornado missile impact and tornado wind loads (per the missile spectrum requirements of UFSAR Table 3.5-11), seismic loads and tornado depressurization loads, as required.

The fundamental design methodology for this new type (for Seabrook) of tornado barrier uses an elastic-plastic deformation approach that absorbs energy from worse-case tornado missile impacts. Bechtel Topical Report BC-TOP-9A (Ref. 18), and other appropriate guides and standards, are used for the barrier design. The full range of required tornado missiles for the Seabrook Station from UFSAR Table 3.5-11 are resisted, and at all postulated trajectories of impact, as required by UFSAR Section 3.5.1.4. Consistent with UFSAR Section 3.5.3, both local barrier perforation and overall response of the barrier are considered in the design.

The structural design evaluation of the missile barrier includes both local effects and overall structural response analyses of the barrier subjected to a horizontal or vertical impact by design basis tornado missiles. For local analysis, the resistance of the steel portions of the barrier against missile perforation is evaluated using the provisions of BC-TOP-9A (Ref. 18). For the gate (door) portion of the barrier, horizontal missile perforation (particularly for smaller missiles, such as the 1" diameter steel rod) is resisted by a combination of ½" thick steel plate and a grid-work webbing of 12" wide fabric straps (2 double plies vertically and 1 horizontal single ply sandwiched between the vertical plies for a total of 5 plies) located behind the steel plate. Provisions of BC-TOP-9A (Ref. 18, section 2.3) are used to calculate the residual velocity of missiles that (by design analysis) will perforate the exterior ½" steel plate of the gate. Energy balance methods of BC-TOP-9A (Ref. 18, section 3.3.2) are then used to ensure that the relatively flexible fabric webbing has sufficient strain energy capacity to resist the remaining kinetic energy (demand) of the worst case missile such that the missile is fully stopped by the webbing. The barrier calculation for this EC (Ref. 25) indicates that substantial strain energy capacity design margin exists for the fabric webbing members. The larger tornado missiles (primarily the automobile missile) engage substantially more barrier sub-members (via a combination of ½" thick steel plate, multiple fabric straps and several 2" diameter horizontal polyethylene fiber cables).

For overall structural response of the barrier due to missile impact, maximum ductility demands of barrier members are determined based on the provisions of BC-TOP-9A (Ref. 18). Unlike conventional elastic or plastic steel design using Parts 1 or 2, respectively, of the AISC code specification, the methodology utilized for the evaluation and design of this new/different type of tornado missile barrier for Seabrook is based on the maximum calculated ductility ratio (maximum deformation normalized by yield deformation), which is a measure of the overall barrier's response and structural damage. Ductility demand associated with a tornado missile striking various components and locations of the missile barrier is quantified using methods in accordance with BC-TOP-9A (Ref. 18). The quantified maximum ductility demand is subsequently compared with permissible ductility ratios to ensure that the elastic-plastic deformation of barrier members does not exceed pertinent allowables. Limiting ductility ratios for barrier members to established code/standard maximum limits ensures that structural damage is maintained within acceptable limits such that a barrier breach by the governing (worst-case) missile is precluded.

Per NUREG 800, NRC SRP 3.5.3, the maximum allowable ductility ratios for steel barrier members are given in ANSI/AISC N690-1994 (Ref. 23). More specifically, the allowable ductility ratios utilized in the qualification of the Barrier 1 missile barrier for this EC are based on Table Q1.5.8.1 of ANSI/AISC N690-1994. Both ANSI/AISC N690-1994 (Table Q1.5.8.1) and BC-TOP-9A (Table 4-4) permit a maximum ductility ratio of 20 for flexure of beams that are proportioned to preclude lateral and local buckling. This is used for the Barrier 1 design since all the HSS tube structural steel members used in the barrier are compact sections and lateral bracing is provided to preclude lateral buckling of the members. For the Barrier 1 design, the steel column members were conservatively designed elastically (i.e., allowable maximum ductility ratio of 1.0; see section 6.5 of Ref. 11). Thus this barrier design is more conservative than the requirements of Bechtel Topical Report BC-TOP-9A, which permits a ductility ratio of 1.3.

The new barrier also includes an embedded reinforced concrete foundation. The foundation is essentially a large inertial mass designed to adequately resist the large governing dynamic loads from tornado missile impact onto the above-ground portion of the barrier. The foundation is designed to meet minimum temperature and shrinkage reinforcing steel requirements of ACI 349-06 (Ref. 27), which are also consistent with the ACI 318-71 (Ref. 26) design code of record for the Seabrook Station for concrete structures. The Barrier 1 missile barrier utilizes energy absorbing members, therefore steel design code ANSI/AISC N690-1994 (Ref. 23) is also utilized, and supplemented with the AISC 13th Edition (Ref. 28).

Tornado doors are steel barriers designed to resist tornado-generated missiles in combination with tornado wind pressures (Reference 21). Energy and momentum principles (Reference 15) were used to determine the maximum equivalent static loads due to missile impact. The 12" diameter steel pipe missile was found to produce the worst case load which was subsequently used in all analyses. Linear elastic finite element analyses of the total door system, including hinges, determined deflections, forces and stresses up to first yield in the door. A yield line analysis was performed to assess maximum capacity allowing for elasto-plastic behavior. Ductility up to 20 was considered acceptable (Reference 15). The hinges and latches were evaluated for maximum shear and axial forces.

3.5.4 References

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

a. Introduction

General Design Criterion 4 of Appendix A to 10 CFR 50 requires that structures, systems and components important to safety be protected against the dynamic effects of piping failures. This section discusses the design bases and design measures used to ensure that all essential structures, systems and components located inside and outside the reactor containment, including the components of the reactor coolant pressure boundary, have been adequately protected against the effects of possible blowdown jet and reactive forces and pipe whips resulting from postulated rupture of piping located both inside and outside of containment.

The required information is furnished in two sections:

1. Subsections 3.6(B).1 and 3.6(B).2 and respective subsections address all piping systems inside and outside containment, exclusive of the reactor coolant loop piping.

2. Section 3.6(N) and subsections, which have been furnished by the NSSS supplier, address only the reactor coolant loop piping inside the reactor containment and the loops' support system.

The criteria used to identify high energy piping and assure the protection of essential equipment from the potential failure of nearby piping systems are per the NRC's Auxiliary Systems Branch (ASB) Technical Position ASB 3-1. The criteria employed to define and locate postulated piping breaks/cracks and their associated effects are the NRC's Mechanical Engineering Branch (MEB) Technical Position MEB 3-1. The MEB 3-1 break postulation criteria supersede the more conservative criteria initially used for piping systems outside containment (ASB 3-1) and inside containment (Regulatory Guide 1.46). For the reactor coolant loop, Westinghouse Topical Report, WCAP-8082, "Pipe Break for the LOCA Analysis of the Westinghouse Primary Coolant Loop," provides the original criteria for postulating breaks in the RC loop. The basis for eliminating eight of these postulated pipe breaks in the RC loop is provided in Reference 3. The Seabrook Station coolant system piping is consistent with the design considered in WCAP-8082, which has been approved by the NRC staff.

b. Definitions

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

- Maximum operating temperature exceeds 200°F
- Maximum operating pressure exceeds 275 psig.

Moderate Energy Fluid Systems or Lines - Fluid system or lines which during normal plant conditions, are either in operation or maintained pressurized above atmospheric pressure under conditions where both the following conditions are met:

- Maximum operating temperature is 200°F or less
- Maximum operating pressure is 275 psig or less.

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

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

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

Postulated Piping Failure - Longitudinal and circumferential breaks in high energy fluid system piping and through-wall leakage cracks in moderate energy fluid system piping postulated according to the provisions of Subsection 3.6(B).2 below.

S_A - Allowable stress range for thermal expansion as defined in subarticle NC3600 of the ASME Code, Section III, 1971 Edition, with Addenda up to and including Winter 1972.

S_h - Allowable stress at maximum temperature.

S_m - Design stress intensity defined in subarticle NB-3600 of the ASME Code.

Single Active Component Failure - Malfunctions or loss of function of a component of an electrical or fluid system. A failure of an active component of a fluid system is not considered to include loss of component structural integrity. The direct consequences of a single active component failure are considered to be part of the single failure.

Terminal Ends - Extremities of piping runs that connect to structures, components (vessels, pumps, valves) or pipe anchors that act as rigid constraints to thermal expansion. A branch connection to a main piping run is a terminal end of the branch run.

Intersections of runs of comparable size and fixity need not be considered terminal ends when so justified by the analysis. Terminal ends, for the purpose of postulating breaks, are selected at points located immediately outside or beyond the required pipe whip restraints located inside and outside containment at penetration areas. For lines that are pressurized by virtue of a connection into a pressurized, normally operating high energy line, the connected line is designated as high energy back to the first normally closed automatic or manual valve or to the first in-line normally closed check valve. In a piping run of this type, the terminal end is considered to be at the piping connection to the closed valve. In the event of valve leakage during normal plant operation, portions beyond this valve might become pressurized. A pipe break in these portions is not considered since it would not produce any sustained pipe whipping or jet impingement due to a lack of an energy reservoir.

Five Degree Restraint - A device which restrains the pipe in such a way that only axial loads can be transmitted past the restraint.

3.6(B).1 Postulated Piping Failures in Fluid Systems

3.6(B).1.1 Design Bases

a. Equipment Potentially Susceptible to Effects of Piping Failure

Systems and components important to plant safety or shutdown (referred to as essential systems and components), located proximate to high or moderate energy piping systems, which are potentially susceptible to the consequences of piping systems breaks and cracks, are listed in Table 3.6(B)-1. The identification of this equipment is related to predetermined piping failure locations, determined by the methodology discussed in Subsection 3.6(B).2.

Table 3.9(B)-27 and Table 3.9(N)-11 tabulate all active valves whose function must be unimpaired in the event of a piping failure.

Figure 3.6(B)-1, Figure 3.6(B)-2 and Figure 3.6(B)-5, Figure 3.6(B)-6, Figure 3.6(B)-7, Figure 3.6(B)-8, Figure 3.6(B)-9, Figure 3.6(B)-10, Figure 3.6(B)-11, Figure 3.6(B)-12, Figure 3.6(B)-13, Figure 3.6(B)-14, Figure 3.6(B)-15, Figure 3.6(B)-16, Figure 3.6(B)-17, Figure 3.6(B)-18, Figure 3.6(B)-19, Figure 3.6(B)-20, Figure 3.6(B)-21, Figure 3.6(B)-22, Figure 3.6(B)-23, Figure 3.6(B)-24, Figure 3.6(B)-25, Figure 3.6(B)-26, Figure 3.6(B)-27, Figure 3.6(B)-28, Figure 3.6(B)-29, Figure 3.6(B)-30, Figure 3.6(B)-31, Figure 3.6(B)-32 show the locations of pipe whip restraints with respect to the postulated pipe ruptures and relative locations of potentially affected essential components.

The limiting acceptable conditions for, and the measures taken to protect the essential systems and components, are discussed in the pipe rupture analysis summary in Appendix 3A.

b. Design Criteria for Protection Against Piping Failures

The following criteria were used as guidelines during the station design to assure the protection of essential equipment from potential failure of nearby piping systems:

1. Piping Systems Containing High Energy Fluids

(a) Piping systems are isolated by adequate physical separation, and remotely located from essential systems and components required to shut down the reactor safely and maintain the station in a cold shutdown condition.

(b) Where isolation by remote location is considered impractical, piping systems, or portions of the systems, are enclosed within structures suitably designed to protect adjoining essential systems and components from postulated piping failures within the enclosure.

(c) Where both isolation by remote location and enclosure in protective structures are considered impractical, the piping systems or portions of the systems are provided with restraints and protective measures so that the operability and integrity of the structures, safety systems and components would not be impaired.

(d) Protective enclosures for the piping systems are designed as seismic Category I structures capable of withstanding the combined effects of a postulated pipe break, the dynamic effects of pipe whipping, the jet impingement forces and the compartment pressurization resulting from discharging fluids in combination with the specified seismic event of the Safe Shutdown Earthquake and normal operating load.

(e) Piping systems containing high energy fluids are designed so that the effects of a single postulated pipe break will not initiate unacceptable failures of other pipes or components. In addition, any systems, or portions of systems, that are designed to mitigate the consequences of a postulated pipe failure and place the reactor in the cold shutdown condition, are provided with design features that will ensure the performance of their safety function, assuming a single active component failure.

(f) For a postulated pipe failure, an escape of steam, water and heat from structures enclosing the piping shall not preclude: (1) the accessibility to surrounding areas important to the safe control of reactor operations, (2) the habitability of the control room, (3) the ability of instrumentation, electric power supplies, and components and controls to initiate, actuate and complete a safety action. In this regard, a loss of redundancy is considered permissible, but not the loss of function.

(g) The design measures used for the protection of structures, systems, and components important to safety will not prevent in-service examinations of ASME Class 2 and 3 pressure-retaining components, as required by the rules of ASME B&PV Code, Section XI, "In-Service Inspection of Nuclear Power Plant Components."

2. Piping Systems Containing Moderate-Energy Fluids

(a) Piping systems containing moderate-energy fluids are designed to comply with the criteria applied to high-energy fluid piping systems, as stated above, except that the piping is postulated to develop a limited-size through-wall leakage crack instead of a pipe break.

(b) For each postulated leakage condition, design measures are provided that will provide protection from the effects of the resulting water spray and flooding.

3. <u>Exceptions</u>

Measures for protection against pipe whipping or jet impingement resulting from the breaks postulated in Subsection 3.6(B).2 are not provided for piping where any of the following applies:

(a) Piping is physically separated or isolated from any essential system or component necessary for plant safety or shutdown by means of barriers, or is restrained from whipping by plant design features such as encasement.

(b) The broken pipe cannot cause unacceptable damage to any essential system or component.

(c) The energy associated with the whipping pipe can be demonstrated to be insufficient to impair to an unacceptable level the safety function of an essential system or component. For example, a whipping pipe is considered unable to rupture an impacted pipe of equal or larger nominal pipe size and equal or heavier wall thickness.

3.6(B).1.2 <u>Description</u>

High energy lines located in structures housing components essential for safe plant shutdown are listed in Table 3.6(B)-2.

Relative to possible dynamic effects of pipe failure in the Seabrook plant layout, essential systems and components are protected from the dynamic effects of rupture of high energy piping primarily by separation and redundancy. Routing of high energy lines has been arranged to provide the maximum amount of protection by using plant structural elements, such as wall or columns, and routing the high energy lines as far as practicable from essential components. In cases where separation is not possible, pipe whip restraints are used to prevent uncontrolled whipping of the high energy piping. Compartments of primary interest are the containment structure, the main steam and feedwater pipe tunnels, and the Containment Enclosure Building and its attached compartments.

In the case of the control room, there are no high energy lines in the area which could affect habitability as a result of pipe whip. The main steam and feedwater lines on the pipe bridge are separated from the control room by the seismic Category I Control Building wall, which has been reinforced to protect the control room environment from postulated breaks in, or whip loads from, the main steam and feedwater lines. Control room habitability systems are discussed in Section 6.4.

The high energy lines outside containment whose breaks or cracks could have the greatest effect on environment within the structures housing components essential for safe plant shutdown are listed below:

a. Primary Auxiliary Building

- Steam generator blowdown lines
- Auxiliary steam and condensate lines
- Chemical and volume control system letdown line
- Hot water heating lines

b. Fuel Storage Building

- Hot water heating lines

c. Containment Enclosure and Connected Buildings

- Hot water heating lines

d. Main Steam and Feedwater Pipe Chase

- Main steam lines
- Feedwater lines

e. Diesel Generator Building

- Hot water heating line

f. Control Building

- Hot water heating line

g. Emergency Feedwater Pumphouse

- Hot water heating line

h. Service Water Pumphouse

- Hot water heating lines

3.6(B).1.3 <u>Safety Evaluation</u>

Appendix 3A summarizes the results, including environmental, of the failure modes and effects analysis of breaks/cracks in high and moderate energy piping systems in each of the structures housing essential components. Appendix 3A verifies that the consequences of failures of high and moderate energy lines will not affect the ability of the plant to be shut down safely. The analysis considered the effects of single active component failures occurring in required systems concurrent with the postulated event.

The potential effects of internally and externally generated missiles are discussed in Section 3.5.

Pressure rise analyses of structures and compartments due to piping breaks are discussed in Section 6.2 and Appendix 3I.

3.6(B).2 <u>Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping</u>

This section describes the design bases for locating postulated breaks and cracks in piping situated both inside and outside of containment, the procedures used to define the jet thrust reaction at the break or crack location, and the jet impingement loading on adjacent safety-related structures, equipment, systems and components.

3.6(B).2.1 <u>Criteria Used to Define Break and Crack Location and Configuration</u>

The criteria employed for defining break and crack locations and configurations in primary loop piping inside containment is discussed in Subsection 3.6(N).2.1. This section discusses all other plant piping.

The criteria are provided for those high and moderate energy piping systems for which separation or enclosure cannot be achieved.

a. High Energy Piping

1. ASME Section III Code Class 1 Piping

Breaks were postulated to occur at the following locations in each piping run or branch run:

(a) Terminal ends

(b) At all intermediate locations where the maximum stress range, derived on an elastically calculated basis under the loadings associated with operating plant conditions and an operating basis earthquake, as calculated by Equation 0 and either Equations 12 or 13 of ASME Code NB-3650 exceeds $2.4S_m$, where S_m is the allowable design stress intensity value.

(c) At intermediate locations where the cumulative usage factor exceeds 0.1.

2. ME Section III Code Class 2 and 3 Piping

Breaks were postulated to occur at the following locations for each piping run or branch run which does not penetrate the containment:

(a) Terminal ends

(b) Any intermediate locations between terminal ends where stresses as calculated by Equations (9) and (10) of NC/ND-3652 derived by elastic methods under the loadings associated with operational plant conditions and an operating basis earthquake exceed 0.8 $(1.2\ S_h + S_A)$. Initial break postulations using the conservative ASB 3-1 (outside containment) and R.G. 1.46 (inside containment) criteria of 0.8 $(S_h + S_{A)}$ are conservative.

See below for piping penetrating the containment.

3. Nonnuclear Piping

Breaks in nonnuclear piping were postulated at the following locations in each piping run or branch run:

(a) Terminal ends

(b) Each structural discontinuity (elbows, tees, reducers, Valves).

4. Piping Penetrating Containment

All piping penetrating the containment is ASME Section III, Code Class 2. All high energy, high temperature lines penetrating containment make use of integrally forged flued heads. A detailed discussion of the design of these flued heads is given in Reference 1.

For main steam and feedwater piping penetrating containment, no breaks were postulated between the first whip restraint inside the containment and the five-degree restraint outside containment, since the following conditions are met:

(a) The maximum stresses, as calculated by the sum of equations (9) and (10) in paragraph NC-3652 of Section III of the Code (for piping design, the applicable Code edition is the 1971 Code, with addenda up to and including Winter 1972), considering normal and upset conditions and an OBE event, do not exceed $0.8(1.2\ S_h + S_a)$.

(b) The maximum stress, as calculated by equation 9 of paragraph NC-3652 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of the piping, does not exceed $1.8\ S_h$.

(c) The number of circumferential and longitudinal weld points in piping have been minimized.

(d) The length of those portions of piping have been reduced to the minimum practical.

(e) The design of pipe restraints and anchors have not generally required welding directly to the outer surface of the pipe. Where anchors or restraints were needed, forgings were used to avoid welding to the surface of the pipe. Where lugs were used for riser clamps, a detailed analysis was made to assure compliance with stress limits stated above.

To assure the protection of essential equipment from the jet impingement and environmental effects of high energy line breaks, longitudinal breaks were postulated (but not concurrently) in the main steam and feedwater lines in the pipe tunnels outside containment. Each postulated longitudinal break was considered to have a cross-sectional area of one-square foot, and was assumed to occur at a location that has the greatest effect on essential equipment.

Since the primary reason for postulating the one-square foot break is for the protection of essential equipment, jet impingement on nearby structures was not evaluated. Environmental effects (pressure, temperature) were evaluated for pipe tunnel structures.

Lug attachments welded to Class 2 and 3 pipes are qualified by a procedure whose equivalent methodology is more conservative than that presented in Code Case N-318-1.

Local stress levels in the pipe resulting from applied lug loads are obtained by multiplying the nominal stress in the lug at the lug/pipe interface by the appropriate B or C index (as defined in Code Case N-318-1) for each individual loading condition. The local stresses are superimposed upon the general pipe stress as determined from program ADLPIPE to establish the total stress level in the pipe for that loading condition.

Loading conditions required to be considered for Plant Normal, Plant Upset, Plant Emergency, and Plant Faulted Operating Condition are defined (per appropriate Updated FSAR section), and total stress in the pipe is obtained from summing the stresses for each individual loading condition that must be considered.

Local stress levels determined using B indices are added to the general stress levels from ADLPIPE and this sum is compared against allowable limits to demonstrate structural integrity. For the pipe wall, local stress levels determined using C indices are added to the general stress levels from ADLPIPE, and this sum is compared against the allowable range of stress ($S_h + S_a$).

The terminal ends of these portions of piping are considered to originate at a point adjacent to the restraints located inside and outside containment which are:

(a) Located reasonably close to the isolation valve

(b) Capable of withstanding the loadings resulting from a postulated pipe rupture beyond this portion of the piping, such that neither valve operability nor the leaktight integrity of the containment is impaired.

Details of typical containment piping penetrations showing location of process pipe welds, anchorage and points of discontinuity are shown in Figure 3.6(B)-3 and Figure 3.6(B)-4.

Augmented in-service inspection of piping in containment penetration areas, along with inservice inspection of Code Class 2 components, is discussed in Section 6.6.

5. With the exception of piping penetrating Containment (Section 4. above) leakage cracks are postulated in ASME Section III Code Class 1 piping where the stress range by Equation 10 of Paragraph NB-3653 exceeds $1.2S_m$ and in Class 2 and 3 or nonsafety class piping where the stress by the sum of Equation (9) and (10) of Paragraph NC/ND-3652 exceeds $0.4(1.2\ S_h + S_A)$. Nonsafety class piping which has not been evaluated to obtain similar stress information has cracks postulated at locations that result in the most severe environmental consequence.

b. Moderate Energy Piping

Through-wall leakage cracks are postulated to occur in seismic Category I and nonnuclear fluid system piping located within or outside and adjacent to protective structures with the following exceptions:

1. Fluid system piping between isolation valves, provided they meet the requirements of ASME Section III, subarticle NE-1120, and are designed so that the maximum stress range does not exceed 0.4 (1.2 S_h + S_A) for ASME Class 2 piping.

2. Fluid system piping located in an area in which a break in a high energy system is postulated, provided a break in a moderate energy fluid system does not result in a more limiting condition than the break in the high energy system.

3. Seismic Category I fluid systems in which the maximum stress range in Class 2 or Class 3 or nonnuclear piping is less than 0.4 (1.2 S_h + S_A).

Moderate and high energy piping cracks per a.5 above were postulated to occur in those locations that result in the maximum effects from spraying and/or flooding.

Through-wall leakage cracks were postulated instead of breaks in the piping of those systems that qualify as high energy fluid systems for only short operational periods, but qualify as moderate energy fluid systems for the major operation period. These systems include containment spray, safety injection and residual heat removal.

An operational period is considered short if the fraction of time that the system operates within the pressure-temperature limits specified for a high energy system is 2 percent or less of the time that it operates as a moderate energy fluid system.

c. Type of Break

The following types of breaks and cracks were postulated to occur in high energy and moderate energy piping as described below:

1. High Energy Piping

(a) Circumferential breaks were postulated to occur in high-energy piping larger than one inch nominal pipe size. Circumferential breaks are presumed to occur at right angles to the axis of the pipe, to completely sever the pipe within one millisecond and to separate the ends of the pipe to permit a flow area equal to the flow area of the pipe, except where pipe whip restraints function to limit pipe separation. See Subsection 3.6(N).2.1 for exception for RCS piping.

b) Longitudinal splits were postulated to occur in high-energy piping four inches or larger nominal pipe size. The area of the longitudinal split was assumed to be equal to the flow area of the pipe, and the split was assumed to be parallel to the axis of the pipe. Jet impingement analysis was based on a rectangular break $2D_i$ long by $\pi D_i/8$ wide where D_i = pipe inside diameter. Breaks were oriented (but not concurrently) at two diametrically opposed points on the piping circumference such that the jet reactions produce out-of-plane bending of the piping configuration.

(c) Certain longitudinal break orientations were excluded on the basis of the state of stress at the location considered. Specifically, where the maximum stress range in the axial direction is at least one and a half times that in the circumferential direction considering upset plant conditions, then only a circumferential break was postulated.

(d) Longitudinal breaks were not postulated to occur in piping at terminal ends.

2. Moderate Energy Piping

   Through-wall leakage cracks were postulated to occur in moderate energy piping larger than one inch nominal pipe diameter, and to have openings of one-half pipe diameter by one-half the pipe wall thickness.

d. Jet Impingement Force Criteria

The criteria used to evaluate jet impingement forces are described in Appendix 3C, Procedure for Evaluation of Jet Impingement Loads from High Energy Piping Failures. After jet forces imposed on structures or equipment have been determined, the capacity of the structures or equipment to support these loads without damage is investigated using conservative methods. Jet impingement loads are considered to be faulted condition loads and are so evaluated.

3.6(B).2.2 Analytical Methods to Define Forcing Functions and Response Models

This section presents a description of the methods used to define forcing functions and response models for pipe whip analysis. For RC Loop piping, see Subsection 3.6(N).2.2.

a. Forcing Functions

1. Time Dependence

   The normal steady-state operating conditions of the plant were assumed prior to postulating a pipe rupture.

   When circumferential ruptures were postulated, the through-wall crack was assumed to develop across the circumference of the pipe instantaneously, and the ruptured pipe was assumed to separate to the full flow area (e.g., double ended rupture) in one millisecond.

   When longitudinal ruptures were postulated, the time for a longitudinal rupture to open to its maximum area was assumed to be one millisecond.

2. System Friction Loss Dependence

   In calculating forces acting on the piping system, full credit may be taken for any restrictions or line losses between the break and the pressure reservoir(s).

3. Closed-Ended Lines

For the closed end of a line (dead end or normally-closed valves) when it was obvious that the fluid dynamic forces could not be sustained, pipe whip response was not considered.

4. Discharge Coefficient

For flashing or nonflashing flow through circumferential and longitudinal breaks, a discharge coefficient, C_d, of 1.0 was used to determine the flow rate through the break,

$$Q = C_d AV$$

where: Q = flow rate through break

A = break flow area

V = velocity

C_d = discharge coefficient

5. Options

The jet thrust reaction, forcing function at the break locations may be generated from a dynamic fluid system model. However, a simplified approach was used, applying a maximum thrust value defined for discharge of nonflashing liquid or for discharge of saturated or superheated vapor as:

$$T = k (P_o - P_a) A$$

where:

T	=	jet thrust reaction force
k	=	thrust factor (1.26 for steam and flashing fluids and 2.0 for sub-cooled, nonflashing fluids)
P_o	=	fluid pressure in pipe (psig)
P_a	=	ambient pressure outside system (psig)
A	=	break flow area

Limited pipe displacement at the break location, line restrictions, positive pump-controlled flow and the effects of pipe friction may be taken into account to reduce the jet force.

For circumferential breaks, direction of thrust was assumed to be along the centerline of the pipe in a direction opposite the jet flow.

For longitudinal breaks, thrust was assumed in a direction opposite jet flow.

For all breaks, maximum thrust was assumed to occur within 1 millisecond and to be a steady-state condition thereafter.

3.6(B).2.3 Dynamic Analysis Methods to Verify Integrity and Operability

a. Dynamic Analysis Methods

The analysis of a piping system and its restraints under pipe rupture conditions requires consideration of the interaction effects of both piping and restraints. The magnitude and distribution of loadings depend upon such parameters as the restraint load-deflection, gaps between piping and restraint, piping flexibility, break location, etc.

1. Energy-Balance Analysis

In this method, kinetic energy generated during the first quarter cycle movement of the ruptured pipe is imparted to the piping/restraint system through impact and is converted into equivalent strain energy. Deformations of the pipe and the restraint are compatible with the level of absorbed energy. For applications where pipe rebound may occur upon impact of the restraint, an additional amplification factor 1.1 was used to establish the magnitude of the forcing function to determine the maximum reaction force of the restraint after the first quarter cycle response. Amplification factors other than 1.1 may be used if justified by more detailed dynamic analysis. Appendix 3D presents the procedure used for calculating piping/restraint system loads by the energy balance method.

2. Quasi-Static Analysis

In order to satisfy the system capability requirements, a dynamic analysis is the preferred method. If dynamic analysis is not possible or feasible for a piping and restraint system, a quasi-static analysis may be possible if it is shown to give more conservative results.

Two design considerations are required as in the dynamic analysis. The system must be capable of supporting both the dynamic and the steady-state blowdown loads.

If a constant, conservative blowdown force is assumed, the system is independent of the dynamic event occurrence time. Since the dynamic inertia effects are therefore unknown, the load-sharing relationship between the pipe and the restraints, etc., cannot be determined.

The jet force can be represented by a conservatively amplified static loading, and the ruptured system is analyzed statically. The amplification factor that is used to establish the magnitude of the forcing function is a conservative value obtained by comparison with the factors derived from detailed dynamic analysis performed on comparable systems. Appendix 3E presents the procedure used for calculating piping/restraint system loads by the equivalent static analysis method.

b. Design Considerations

Pipe rupture locations and orientation were determined as stated in Subsections 3.6(B).2 and 3.6(N).2. Effects of each rupture were evaluated and, if necessary, whip restraints were located to protect the essential systems or components.

For Code Class 1, 2 or 3 piping, the whip restraints were designed to prevent unrestrained whipping of the piping, but at the same time permit unrestrained thermal movement of the piping.

In some cases, such as on the main steam and feedwater lines in the penetrations and piping tunnel areas, it was appropriate to use pipe whip restraint steel as an extension of the building steel for the attachment of seismic restraints. Wherever this was done, the boundary between PWR steel and ASME Class 2 seismic restraints was defined by showing the PWR steel and the seismic restraints on separate fabrication and installation drawings. All Code Class supports are identified on the drawings as "N-Stamp Items."

After the whip restraints were located, the following information was developed:

- Jet thrust force
- Pipe seismic displacement
    - Pipe thermal displacement
    - Maximum allowable pipe travel
    - Insulation thickness

Minimum gap between pipe and restraint is determined from consideration of 2, 3 and 5 above. Restraint stiffness is determined from 1 and 4. Where the whip restraint is also a seismic restraint, the following values for stiffness were used:

- For piping larger than 8" nominal diameter: 10^6 or 10^7 lb./inch
- For piping from 2½" to 6" nominal diameter: 10^5, 10^6 or 10^7 lb./inch
- For piping up to 2" nominal diameter: 10^4, 10^5 or 10^6 lb./inch.

Analyses of representative piping configurations show that a change in stiffness of one order of magnitude in either direction will not change pipe stresses significantly, so that the designers generally used the lowest values for stiffness in the ranges given, unless pipe deflection is the critical parameter.

In the design of the pipe whip restraints, the energy absorption capacity of the pipe was not considered.

To determine the adequacy of a system, including pipes and restraints, following a postulated pipe rupture accident, two design considerations were evaluated:

1. <u>Dynamic Response</u>

   Upon the occurrence of the postulated pipe rupture, the system of pipe restraints, structure, etc., will respond dynamically to the suddenly applied blowdown thrust, $F_B(t)$. This thrust will move the pipe so that it impacts against the restraint with an impulse equal to the pipe mass times the impact velocity. The product of blowdown thrust, F_B, and the time after this impact until motion ceases, t, will be an additional impulse on the system.

2. <u>Static Equilibrium</u>

   Following the occurrences of the dynamic event (when motion ceases), the system must be able to support the active applied forces (the blowdown thrust). Therefore, the system must satisfy the requirements of a static analysis.

For a conservative static analysis, each component (i.e., pipe, restraint) is capable of supporting the total load (or it is shown which component(s) support the load). When this is done and the components will have the load capacity to support the steady-state blowdown, the system design is considered to be conservative.

c. Piping Design Loading Combinations

Pipes which have been identified in Subsection 3.6(B).1 as those which could cause adverse effects due to pipe movement were provided with means of controlling their motion, if barriers, separation, or some other acceptable method was not used for protection.

1. Adequacy Requirements

To control the motion of pipes, it is necessary that the load on the pipe during the dynamic event be less than the load capacity of the pipe. The dynamic load capacity of the pipe can be determined by test or by a suitable analytical model.

Without testing, the load capacity for analysis is limited to the bending associated with a maximum fiber strain of 50 percent of the ultimate strain of the pipe material. Ultimate strain is defined as the value of strain which corresponds to the maximum stress on the engineering stress-strain diagram. For a given material where there is a range of values due to statistical variation, the guaranteed minimum value of ultimate strain is used.

The second requirement, to insure that the motion of pipes is controlled, is for the moment-carrying capacity of the pipe to be greater than the applied moment after the occurrence of the dynamic event. An ultimate moment, M_u, is defined as the maximum moment that the pipe cross section can support. If the applied moment, M_a, defined as a force times lever arm is numerically smaller than M_u, uncontrolled rotation of the pipe will not occur.

2. Material Properties

Careful consideration was given to the piping material properties used. The rapid loading conditions due to pipe rupture may require the consideration of high strain-rate effects on material behavior, in addition to strain-hardening considerations. Section III of the ASME Code provides tabulations of material properties which may be used for some evaluations. The values of yield strength at temperature, for example, are minimum values for static loadings. In calculating the allowable span distance between restraints, use of minimum values is conservative. In calculating the maximum moment which could be exerted on an anchor point, the use of minimum values would not be conservative. The applied moments, M_a, during the steady-state blowdown will be no greater than 90 percent of the moment capacity of the pipe based on minimum pipe material properties determined from test, applicable specifications, or codes.

3.6(B).2.4 Guard Pipe Assembly and Pipe Whip Restraint Design Criteria

a. Guard Pipe Assembly

Guard pipes are used in the following locations: (a) on the main steam and feedwater lines to prevent pressurization of the containment enclosure in the event of a pipe rupture; (b) on the main steam lines just north of the main steam and feedwater pipe chase to protect the main steam isolation valves from damage due to jet impingement of the pipe chase north wall; and (c) on the main steam line in the pipe bridge area to protect the Control Building wall from jet impingement. The guard pipes in the containment enclosure were designed as a part of the flued head penetrations for the main steam and feedwater lines. A discussion of the design criteria and analysis of the high energy containment penetrations is given in Reference 1. The purpose of the penetration assemblies is to permit penetration of the containment by process pipes without jeopardizing containment integrity. Where they are used as guard pipes, they also serve to prevent overpressurization of the containment enclosure and annulus. No other lines in this area require guard pipes.

In general, all process pipes penetrating containment are seamless. Penetration assemblies for large high temperature lines and steam generator blowdown lines are integrally forged flued head design. Penetration assemblies for cold lines or small lines (under 1" nominal diameter) are seamless pipe welded to flat plate heads which are in turn welded to sleeves anchored in the containment structure. All penetration sleeves are seal welded to the steel containment liner, and leak test channels are provided for periodic testing of containment leak-tightness.

There are no process pipe welds located within the protective assemblies, with the exception of the 2" diameter steam blowdown lines. The process pipe welds for these lines do not require in-service inspection (Reference IWC-1220d of ASME XI).

Moment-limiting restraints have been provided for all penetrations carrying high energy piping to maintain process pipe stress levels below the limits defined in Equation 8 of NC3652 for maximum stress range, considering all upset design transients in combination with OBE.

b. <u>Pipe Whip Restraints</u>

For BOP piping, pipe whip restraints are provided where required to protect essential components, to maintain the motion of the ruptured pipe within controlled limits. The limit of pipe motion is the area within which no essential component would be affected by impact or jet impingement.

The primary function of a pipe whip restraint is to control pipe motion upon the occurrence of a pipe rupture. As used in this context, a restraint is considered to be different from a pipe support. In certain instances, a restraint may also function as a part of the building structure to support a pipe support, cable tray support, conduit support, duct support or any other support or combination of these supports.

Typical whip restraints consist of heavy structural extension of the building structure to the pipe, and a structural box or a series of U-bolts which surround the pipe to restrain lateral motion. Unless the restraint acts also as a support for a thermal or seismic restraint, contact between the pipe and the whip restraint is prevented by means of a suitable air gap. Where it is necessary to reduce pipe impact loads on critical structures, energy-absorbing devices are used between the pipe and the supporting structure. Pipe whip restraints are designed for one-time usage, and as such may be allowed to have greater distortion, plastic deformation, etc., than normally permitted for support design.

In general, for pipe whip restraints, elastic design criteria were used. In cases where the pipe whip restraint was also used for attachment of supports or restraints, it was first designed and analyzed as a whip restraint, and then checked to verify its ability to withstand the other support and restraint loads acting simultaneously.

The basis for the designs and analyses of the pipe whip restraints is SRP 3.6.2 and 3.8.3, as referenced in SRP 6.2.1, paragraph 6.2.1.2 addressing structures which behave linearly under the application of an applied load.

If, as a result of pipe whip load reanalysis, the stress limits given above were exceeded, allowable stresses increased by faulted factors based on those allowed in ASME III, Division I, Appendix F, were applied.

For some pipe whip restraints, certain members were designed to elasto-plastic design criteria. In those members designated to behave elasto-plastically, where the effects of strain-rate, strain-hardening, etc., are included, the permanent strain in these metallic ductile materials was limited to 50 percent of the uniform tensile strain. When a crushable energy-absorbing material was used, the deformation was limited to that corresponding to 75 percent of the crush pad core height.

Where wire ropes were used as part of the pipe whip restraint design, Bethlehem Purple Plus IWRC wire rope material was selected, and design loads in the wire rope were limited to 55 percent of its material minimum breaking strength.

Where energy absorption of the whipping pipe was dependent on elasto-plastic action of a pipe whip restraint member (U-bolts), the material selected exhibited high ultimate strain properties.

For the ASTM A193 GR-B8 material selected for the U-bolts, the ultimate tensile strain rate was limited to 40 percent. For the ASTM A276 Type XM-10 material selected for the U-bolts, the ultimate tensile strain rate was limited to 45 percent.

3.6.(B).2.5 Material to Be Submitted for the Operating License Review

The summary of results of the analyses performed on high and moderate energy piping systems and their restraints to determine the effects of postulated pipe breaks and cracks, as well as the procedures used, are presented in the following appendices:

Appendix 3A	Pipe Break Analysis Summary
Appendix 3C	Procedure for Evaluating Jet Impingement Loads from High Energy Piping Failures
Appendix 3D	Procedure for Calculating Elasto-Plastically Designed Pipe Whip Restraint Loads by the Energy Balance Method
Appendix 3E	Procedure for Calculating Elastically-Designed Pipe Whip Restraint Loads by Equivalent Static Analysis Method.

3.6(B).3 References

1. "Stress Report for High Energy Piping Penetrations for PSNH-Seabrook Station Units 1 and 2," Stress Report No. 9763-325-1, (Calculation 9763-C-01-ST-00-F) Rev. 1 dated September, 1976, United Engineers & Constructors Inc.

2. Moody, F.J., "Prediction of Blowdown Thrust and Jet Forces," Paper No. 69-HT-31, presented at the ASME-AICHE Heat Transfer Conference, Minneapolis, Minnesota, August 3-6, 1969.

3. Final Rule Modifying 10 CFR 50 Appendix A, GDC-4 dated October 27, 1987 [52 FR 41288].

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

This section describes the design bases and protective measures that are used to ensure the containment and its equipment are adequately protected against the dynamic effects caused by postulated rupture of the reactor coolant system piping. Elements of this system are discussed in Section 5.4.

3.6(N).1 Postulated Piping Failures in Fluid Systems outside of Containment

Refer to Subsection 3.6(B).1.

3.6(N).2 Determination of Break Locations and Dynamic Effects Associated with Postulated Rupture of Piping

The design bases for postulated breaks in the reactor coolant system piping are given below. The design bases for the postulated pipe rupture include not only the break criteria, but also the criteria to protect other piping and vital systems from the effects of the postulated rupture.

3.6(N).2.1 Criteria Used to Determine Break and Crack Location and Configuration

In any given piping system, there are a limited number of locations that are more susceptible to failure by virtue of stress or fatigue than the remainder of the system.

The discrete break locations and orientations in the reactor coolant loop were originally derived on the basis of stress and fatigue analysis. These postulated break locations and the methods that were used to determine them are described in Reference 1. An analysis of each individual reactor coolant loop confirmed the break locations defined in Reference 1. Actual seismic loads for the Seabrook site are included in the specific plant reactor coolant loop analysis. Eight of these break locations are eliminated from the plant design basis due to the consideration of the detailed fracture mechanics evaluation in Reference 5. This new design was approved by the NRC in Reference 4. For the remaining breaks, (a) pressurizer surge line nozzle, (b) residual heat removal line nozzle and (c) accumulator line nozzle, the breaks are postulated on terminal end criteria.

At postulated circumferential break locations, the piping is assumed to separate and to allow double-ended flow. Longitudinal breaks are assumed to have an opening area equal to the flow area of the pipe.

3.6(N).2.2 Analytical Methods to Define Forcing Functions and Response Models

Following is a summary of the methods used to determine the dynamic response of the reactor coolant loop associated with postulated pipe breaks in the loop piping.

a. Time Functions of Jet Thrust Force on Ruptured and Intact Loop Piping

To determine the thrust and reactive force loads to be applied to the reactor coolant loop during the postulated loss-of-coolant accident (LOCA), it is necessary to have a detailed description of the hydraulic transient. Hydraulic forcing functions are calculated for the intact reactor coolant loops as a result of a postulated LOCA. These forces result from the transient flow and pressure histories in the Reactor Coolant System. The calculation is performed in two steps. The first step is to calculate the transient pressure, mass flow rates, and thermodynamic properties as a function of time. The second step uses the results obtained from the hydraulic analysis, along with input of areas and direction coordinates, and calculates the time history of forces at appropriate locations (e.g. elbows) in the reactor coolant loops.

The hydraulic model represents the behavior of the coolant fluid within the entire Reactor Coolant System. Key parameters calculated by the hydraulic model are pressure, mass flow rate, and density. These are supplied to the thrust calculation, together with plant layout information, to determine the time-dependent loads exerted by the fluid on the loops. In evaluating the hydraulic forcing functions during a postulated LOCA, the pressure and momentum flux terms are dominant. The inertia and gravitational terms are taken into account in evaluation of the local fluid conditions in the hydraulic model.

The blowdown hydraulic analysis is required to provide the basic information concerning the dynamic behavior of the reactor core environment for the loop forces, reactor kinetics and core cooling analysis. This requires the ability to predict the flow, quality, and pressure of the fluid throughout the reactor system. The MULTIFLEX code (Reference 2) was developed with a capability to provide this information.

The MULTIFLEX code performs a comprehensive space-time dependent analysis of a LOCA and is designed to treat all phases of the blowdown. The stages are: (1) a subcooled stage where the rapidly changing pressure gradients in the subcooled fluid exert an influence upon the Reactor Coolant System and support structures, (2) a two phase depressurization stage, and (3) the saturated stage. See section 3.9(N).2.5 for details of the MULTIFLEX code.

The STHRUST computer program was developed to compute the transient (blowdown) hydraulic loads resulting from a LOCA.

The blowdown hydraulic loads on primary loop components are computed from the equation

$$F = 144A\left(p - 14.7\right) + \left[\frac{\dot{m}^2}{\rho\, g_c\, A^2{}_m\, 144}\right]$$

The symbols and units are:

F	=	Force, lb.
A	=	Area, ft^2
P	=	System pressure, psia
$\dot{m}$	=	Mass flow rate, lb_m/sec
ρ	=	Density, lb_m/ft^3
g_c	=	Gravitational constant = 32.174 $ft\text{-}lb_m/lb_f - sec^2$
A_m	=	Mass flow area, ft^2

In the model to compute forcing functions, the Reactor Coolant Loop System is represented by a similar model as employed in the blowdown analysis. The entire loop layout is represented in a global coordinate system. Each mode is fully described by: (1) blowdown hydraulic information and (2) the orientation of the streamlines of the force nodes in the system, which includes flow areas and projection coefficients along the three axes of the global coordinate system. Each node is modeled as a separate control volume, with one of two flow apertures associated with it. Two apertures are used to simulate a change in flow direction and area. Each force is divided into its x, y, and z components using the projection coefficients. The force components are then summed over the total number of apertures in any one node to give a total x force, total y force, and total z force. These thrust forces serve as input to the piping/restraint dynamic analysis.

The STHRUST code is described in Reference 3.

b. Dynamic Analysis of the Reactor Coolant Loop Equipment Supports

The dynamic analysis of the reactor coolant loop piping for the LOCA loadings is described in Section 3.9(N).

3.6(N).2.3 Dynamic Analysis Methods to Verify Integrity and Operability

a. Protective Measures

The fluid discharged from the ruptured piping will produce thrust and reaction forces in the piping systems. The effects of these loadings are considered in assuring the continued integrity of the vital components and the Engineered Safety Features.

To account for these effects in the design, a combination of component restraints, barriers, and layout is used to ensure that for a loss of coolant, steam or feedwater line break, propagation of damage from the original event is limited, and the components as needed, are protected and available.

b. Criteria for Protection Against Postulated Pipe Breaks in Reactor Coolant System Piping

A loss of reactor coolant accident is assumed to occur for a branch line break down to the restraint of the second normally open automatic isolation valve (Case II in Figure 3.6(N)-1) on outgoing lines and down to and including the second check valve (Case III in Figure 3.6(N)-1) on incoming lines normally with flow. (It is assumed that motion of the unsupported line containing the isolation valves could cause failure of the operators of both valves to function.) A pipe break beyond the restraint or second check valve will not result in an uncontrolled loss of reactor coolant if either of the two valves in the line close. Accordingly, both of the automatic isolation valves are suitably protected and restrained as close to the valves as possible so that a pipe break beyond the restraint will not jeopardize the integrity and operability of the valves. Further, periodic testing capability of the valves to perform their intended function is essential. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Cases I and IV in Figure 3.6(N)-1), a loss of reactor coolant accident is assumed to occur for pipe breaks on the reactor side of the valve.

Branch lines connected to the Reactor Coolant System are defined as "large" for the purpose of these criteria if they have an inside diameter greater than 4 inches up to the largest connecting line, generally the pressurizer surge line. Rupture of these lines results in a rapid blowdown from the Reactor Coolant System, and protection is basically provided by the accumulators and the low head safety injection pumps (residual heat removal pumps).

Branch lines connected to the Reactor Coolant System are defined as "small" if they have an inside diameter equal to or less than 4 inches. This size is such that emergency core cooling system analyses using realistic assumptions show that no clad damage is expected for a break area of up to 12.5 square inches corresponding to 4-inch inside diameter piping.

Engineered Safety Features are provided for core cooling and boration, pressure reduction and activity confinement in the event of a loss of reactor coolant or steam or feedwater line break accident to ensure that the public is protected in accordance with 10 CFR 100 guidelines. These safety systems have been designed to provide protection for a reactor coolant system pipe rupture of a size up to and including a double-ended severence of the reactor coolant system main loop.

To ensure the continued integrity of the vital components and the Engineered Safety Systems, consideration is given to the consequential effects of the pipe break itself to the extent that:

1. The minimum performance capabilities of the Engineered Safety Systems are not reduced below that required to protect against the postulated break;

2. The containment (defined as the containment structure liner and penetrations, and the steam generator shell, the steam generator steam side instrumentation connections, the steam, feedwater, blowdown and steam generator drain pipes within the containment structure) leaktightness is not decreased below the design value, if the break leads to a loss of reactor coolant; and

3. Propagation of damage is limited in type and/or degree to the extent that:

    - A pipe break which is not a loss of reactor coolant will not cause a loss of reactor coolant or steam or feedwater line break.

- A reactor coolant system pipe break will not cause a steam or feedwater system pipe break and vice versa.

(a) Large Branch Lines

Large branch line piping, as defined in Subsection 3.6(N).2.3b, is restrained to meet the following criteria in addition to Items 1 through 3 of Subsection 3.6(N).2.3b for a pipe break resulting in a loss of reactor coolant:

(1) Propagation of the break to the unaffected loops is prevented to assure the delivery capacity of the accumulators and low head pumps.

(2) Propagation of the break in the affected loop is permitted to occur but does not exceed 20 percent of the flow area of the line which initially ruptured. This criterion has been voluntarily applied so as not to substantially increase the severity of the loss of coolant.

(b) Small Branch Lines

In the unlikely event that one of the small pressurized lines, as defined in Subsection 3.6(N).2.3b, should fail and result in a loss-of-coolant accident, the piping is restrained or arranged to meet the following criteria in addition to Items 1 through 3 of Subsection 3.6(N).2.3b.

(1) Break propagation is limited to the affected leg, i.e., propagation to the other leg of the affected loop and to the other loops is prevented.

(2) Propagation of the break in the affected leg is permitted, but is limited to a total break area of 12.5 square inches, (4-inch inside diameter). The exception to this case is when the initiating small break is a cold leg high head safety injection line. Further propagation is not permitted for this case.

(3) Damage to the high head safety injection lines connected to the other leg of the affected loop or to the other loop is prevented.

(4) Propagation of the break to a high head safety injection line connected to the affected leg is prevented if the line break results in a loss of core cooling capability due to a spilling injection line.

c. Protective Provisions for Vital Equipment

In addition to pipe restraints, barriers and layout are used to provide protection from pipe whip and blowdown jet forces.

Some barriers used for protection against pipe whip are: the crane wall serves as a barrier between the reactor coolant loops and the containment liner; in addition, the refueling cavity walls, the operating floor, and the crane wall enclose each reactor coolant loop into a separate compartment, thereby preventing an accident, which may occur in any loop, from affecting another loop or the containment liner; the portion of the steam and feedwater lines within the containment has been routed behind barriers that separate these lines from all reactor coolant piping. The barriers described above will withstand loadings caused by jet forces and pipe whip impact forces.

Other than for the emergency core cooling system lines that must circulate water to the vessel, the Engineered Safety Features are located outside of the crane wall. The emergency core cooling system lines which penetrate the crane wall are routed around and outside the crane wall to penetrate the crane wall in the vicinity of the loop to which they are attached.

It has been demonstrated by Westinghouse Nuclear Energy System tests, that lines hitting equal or larger size lines of the same schedule will not cause failure of the line being hit; e.g., a 1-inch line, should it fail, will not cause subsequent failure of a 1-inch or larger size line. The reverse, however, is assumed to be probable; i.e., a 4-inch line, should it fail and whip as a result of the fluid discharged through the line, could break smaller size lines such as neighboring 3-inch or 2-inch lines. In this case, the total break area is less than 12.5 square inches.

Where restraints on the lines are necessary to prevent impact on and subsequent damage to the neighboring equipment or piping, restraint type and spacing are chosen so that a plastic hinge of the pipe at the two support points closest to the break is not formed. Alternatively, if the layout is planned so that whipping of the two free sections cannot reach equipment or other pipes for which protection is required, then plastic hinge formation is allowed.

As another alternative, barriers are erected to prevent the whipping pipe from impacting on equipment or piping requiring protection. Finally, tests and/or analyses are performed to demonstrate that the whipping pipe will not cause damage in excess of acceptable limits.

Whipping in bending of a broken stainless steel pipe section does not cause this section to become a missile. This design basis has been demonstrated by Westinghouse Nuclear Energy Systems bending tests on large and small diameter, heavy and thin-walled stainless steel pipes.

d. Design Loading Combinations

As described in Section 3.9(N), the forces associated with rupture of piping systems are considered in combination with normal operating loads and earthquake loads for the design of supports and restraints to assure continued integrity of vital components and Engineered Safety Features.

The stress limits for reactor coolant piping and supports are discussed in Section 3.9(N).

3.6(N).2.4 Guard Pipe Assembly Design Criteria

Refer to Subsection 3.6(B).2.4.

3.6(N).2.5 Material to be Submitted at the Operating License Review

a. Table 3.6(N)-1 and Figure 3.6(N)-2 identify the design basis pipe break locations for the main reactor coolant loop.

The primary-plus-secondary stress intensity ranges and the fatigue cumulative usage factors at the design break locations specified in Subsection 3.6(B).2.1 are not tabulated since selection of these terminal end locations is independent of detailed stress and fatigue analysis.

b. Design loading combinations and applicable criteria for ASME Class 1 components and supports are provided in Subsection 3.6(N).2.3d. Pipe rupture loads include not only the jet thrust forces acting on the piping but also jet impingement loads on the primary equipment and supports.

3.6(N).3 References

1. "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," WCAP-8082-P-A, January 1975, (Proprietary) and WCAP-8172-A (Nonproprietary), January 1975.

2. K. Takeuchi, et al., "MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal Hydraulic-Structure System Dynamics," WCAP-8708-P-A, Westinghouse Proprietary Class 2/ WCAP-8709-A, NES Class 3 (Non-Proprietary, September 1977.

3. "Documentation of Selected Westinghouse Structural Analysis Computer Codes," WCAP-8252, April 1974.

4. Final Rule Modifying 10 CFR 50 Appendix A, GDC-4 dated October 27, 1987 [52 FR 41288].

5. "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Seabrook Units 1 and 2," WCAP-10567, June 1984 (Proprietary) and WCAP-10566 (Nonproprietary), June 1984.

3.7(B) SEISMIC DESIGN

3.7(B).1 Seismic Input

This subsection contains a discussion of the input criteria to be used for seismic design of the plant. Items included in the discussion are design response spectra and the basis for their selection, earthquake time-motion records and the basis for their selection, response spectra obtained from time-motion records, and recommended percentages of critical damping to be used for seismic analysis.

3.7(B).1.1 Design Response Spectra

The design response spectra for horizontal and vertical motion, corresponding to the SSE applicable to the site, are presented in Subsection 2.5.2. The spectra amplification ratios for various levels of damping used to establish the design response spectra are based upon values presented in Regulatory Guide 1.60. The OBE response spectra are obtained by multiplying the SSE spectra ordinates by one-half. The duration of the earthquake is estimated at 10 to 15 seconds.

There are no existing earthquake records pertinent to the Seabrook Station site.

3.7(B).1.2 Design Time History

The three components of artificial time-history motion corresponding to the SSE are shown on Figure 3.7(B)-1, Figure 3.7(B)-2, and Figure 3.7(B)-3. These figures show the two horizontal motions and the one vertical motion, respectively. The components of the artificial time-history corresponding to the OBE are obtained by multiplying the ordinates of Figure 3.7(B)-1, Figure 3.7(B)-2, and Figure 3.7(B)-3 by one-half (1/2). The earthquake motion was generated by super-imposing sinusoidal waves of many frequencies. Phase angles of the sinusoidal waves are randomly chosen. The resulting wave form is then multiplied by a trapezoidal intensity function to cause the time variation of intensity. All artificial time histories used in the analyses are base-line corrected.

Response spectra obtained from earthquake time history are shown on Figure 3.7(B)-4, Figure 3.7(B)-5, Figure 3.7(B)-6, Figure 3.7(B)-7, Figure 3.7(B)-8, Figure 3.7(B)-9, Figure 3.7(B)-10, Figure 3.7(B)-11, Figure 3.7(B)-12, Figure 3.7(B)-13, Figure 3.7(B)-14, Figure 3.7(B)-15, Figure 3.7(B)-16, Figure 3.7(B)-17, Figure 3.7(B)-18, Figure 3.7(B)-19 and Figure 3.7(B)-20. Included on each figure is the corresponding design response spectra discussed in Subsection 3.7(B).1.1. The response spectra obtained from the time-motion record envelope the design response spectra for the period range of 0.03 seconds to 4.0 seconds. The response spectra have been computed using a method based on the exact solution of the governing differential equation for a single degree of freedom oscillator with viscous damping. To ensure that the response spectra are sufficiently accurate, they were calculated at a set of discrete values for period, T, forming a geometric progression, i.e.,

$To, T_o r, T_o r^2, \ldots T_o rn\text{-}1$

$r \quad = 1.02$

$T_o \quad = 0.03$ seconds

This ratio corresponds to a period interval varying from 0.0006 seconds at a period of 0.03 seconds to a period interval of 0.01 seconds at a period of 0.50 seconds.

3.7(B).1.3 Critical Damping Values

The percentages of critical viscous damping used for the seismic analysis of Category I structures, systems, and components are based on recommendations presented in Regulatory Guide 1.61. These percentages, which account for stress level as well as type of construction or fabrication, are summarized in Table 3.7(B)-1.

For seismic piping analysis, an alternative to Regulatory Guide 1.61 may be used. These values are shown graphically in Figure 3.7(B)-38.

For the Cable Raceway System, an alternative to Regulatory Guide 1.61 may be used. Critical damping levels may be a maximum of 20 percent for input acceleration levels of 0.35g and greater for OBE and SSE conditions. In cases where input accelerations are between 0.1g and 0.35g, the critical damping values may be interpolated between 7 percent and 20 percent, respectively.

3.7(B).1.4 Supporting Media for Category I Structures

All seismic Category I structures are founded on sound bedrock or on engineered backfill extending to sound bedrock. Engineered backfill was also placed around all seismic Category I structures.

The bedrock at the site is uniform, competent, and nonfragmented. Engineering properties of the bedrock measured in both the field and the laboratory are presented in Subsection 2.5.4.2a.

The engineered backfill consists of either fill concrete, backfill concrete, offsite borrow, tunnel cuttings, or sand-cement. Properties of the engineered backfill materials are described in Subsection 2.5.4.5. The type of engineered backfill used beneath all seismic Category I structures was fill concrete, except for safety-related electrical duct banks, five electrical manholes, and the service water pipes, which were founded on offsite borrow or tunnel cuttings, as shown in Table 2.5-20.

Identification of the safety-related electrical manholes founded on offsite borrow or tunnel cuttings, the depths of offsite borrow or tunnel cuttings over the bedrock under these particular manholes, the widths of their structural foundations and the total structural height are summarized below:

Manhole Numbers	Depths of Soil over Bedrock (ft)	Widths of Structural Foundations (ft)	Total Structural Height (ft)	Supporting Material
W13/W14	6-12	18 x 18½	9½	Offsite Borrow
W15/W16	6-12	18 x 18½	9½	Offsite Borrow
W19/20	15	23½ x 23½	12	Tunnel Cuttings
W29/W30	14	19 x 22½	15	Offsite Borrow
W33/W34	18	18 x 18½	12	Offsite Borrow

All manholes are fully embedded.

The values of shear modulus, G, and shear wave velocity, v_s, for both the offsite borrow and tunnel cuttings used for the analyses of the manholes are discussed in Subsection 2.5.4.7.

3.7(B).2 Seismic System Analysis

This subsection contains a discussion of the seismic analyses performed for seismic Category I structures and systems. Included in the discussion are the methods of seismic analysis used, the criteria used for mathematically modelling the structures and systems, the assumptions made in the analyses, and the effects considered.

3.7(B).2.1 Seismic Analysis Methods

The seismic response of Category I structures, systems and components has been determined from suitable elastic dynamic analyses. The results of these analyses are used for the design of seismic Category I structures, systems and components, and are input for subsequent dynamic analyses.

Two methods of seismic system analysis were used for seismic Category I structures: (1) the modal analysis response-spectrum method and (2) the mode-superposition time-history method. The time-history method was used to determine the dynamic response necessary to obtain amplified response spectra for component design. The input forcing functions (the time history of ground motion) are shown graphically in Figure 3.7(B)-1, Figure 3.7(B)-2 and Figure 3.7(B)-3. The time history shown on Figure 3.7(B)-1 is used in both horizontal directions. The peak acceleration is 0.25g for the SSE and 0.125g for the OBE. Design response spectra for the response-spectrum method are shown in Section 2.5.

The mathematical models used for the seismic Category I structures are typically lumped masses connected by linear elastic springs. Each structure, then, is described by a finite number of degrees-of-freedom chosen to represent the principal overall behavior of the system. The modelling is described in Subsection 3.7(B).2.3 in more detail. The number of masses or degrees-of-freedom included in the analysis is determined by requiring the total degrees-of-freedom to exceed twice the number of significant modes with frequencies less than 33 Hz. Up to four degrees-of-freedom were considered for each mass point, three translation and one torsion. The three orthogonal directions were run separately, and results were combined by the grouping method in accordance with Regulatory Guide 1.92.

All significant modes with frequencies up to 50 Hz were used in analyses for both local and overall effects.

The effects due to inertial characteristics of fluid contained within a structural component were considered in the analysis by techniques described in Reference 1. No soil-structure interaction effects were involved because of the rock siting.

The lumped mass mathematical models for representative seismic Category I structures, including the containment, are shown in Figure 3.7(B)-21, Figure 3.7(B)-22, Figure 3.7(B)-23, Figure 3.7(B)-24 and Figure 3.7(B)-25. Subsection 3.7(B)2.3 describes these models in more detail. Table 3.7(B)-2, Table 3.7(B)-3, Table 3.7(B)-4, Table 3.7(B)-5, Table 3.7(B)-6, Table 3.7(B)-7 and Table 3.7(B)-8 are the property tables for the indicated models.

Relative displacements between supports of structures are zero due to the base fixity. Maximum displacements throughout the structure (relative to the fixed base of the model) were computed for use in component analysis, as described in Section 3.7(B).3.

The description of the analysis methods used for seismic Category I systems and components is provided in other sections.

3.7(B).2.2 Natural Frequencies and Response Loads

The seismic analyses of Category I structures are based upon the time history modal superposition and response spectra normal mode methods using idealized lumped mass models of the individual structures. These methods of analyses use the natural frequencies, mode shapes and appropriate damping coefficients of the system. The system frequencies, modes shapes and structural response are obtained using the STARDYNE computer program and the models described in Subsections 3.7(B).2.1 and 3.7(B).2.3. Numerical results are tabulated in this subsection for the following representative seismic Category I structures: Containment Building, Primary Auxiliary Building, Control and Diesel Generator Building, and Fuel Storage Building. Table 3.7(B)-5, Table 3.7(B)-6, Table 3.7(B)-9 Table 3.7(B)-10, Table 3.7(B)-11, Table 3.7(B)-12, Table 3.7(B)-13, Table 3.7(B)-14, Table 3.7(B)-15, Table 3.7(B)-16, Table 3.7(B)-17, Table 3.7(B)-18, Table 3.7(B)-19, and Figure 3.7(B)-20 list natural frequencies, periods, mode classification, nodal accelerations, nodal displacements and peak forces for these buildings.

3.7(B).2.3 Procedures Used for Analytical Modeling

The seismic analyses of Category I structures are based upon dynamic analyses using idealized three-dimensional lumped mass models of the physical structures. The inertial properties of the models are characterized by the lumped mass, eccentricity of the mass, and the torsional mass moment of inertia at each mass point of the model. The locations of lumped masses or nodes are selected at floor slabs, at changes of cross-sectional area and at intermediate points, such as locations of equipment, etc. The number of dynamic degrees-of-freedom are chosen to exceed twice the number of significant modes with frequencies less than 33Hz as a minimum. The concentrated masses are connected by weightless elastic beams representing the resisting structural members between mass points. Torsion is accounted for by rigidly offsetting the centers of mass from the centers of rigidity. Floor slabs are assumed to be rigid in their own plane, and the stiffness of shear walls is appropriately considered taking openings into account. The stiffness properties of the models are characterized by the cross-sectional area, moment of inertia, shear shape factor, torsional constant, and Young's Modulus of Elasticity and Poisson's Ratio. The torsional stiffness of the structure is evaluated using the lateral stiffness of the resisting elements between mass points and the square of their distance from the center of resistance for a particular story. The stiffnesses of a particular story in the model are combined into a pseudo-elastic beam located at the center of resistance of the resisting elements.

All seismic Category I structures, with the exception of some electrical manholes and ductbanks (see Subsection 3.7(B).2.4), are supported on competent bedrock or concrete fill over bedrock. All mathematical models are, therefore, fixed against translation and rotation at their bases. The elevation of the point-of-fixity of the mathematical model is determined as follows:

a. Lowest elevation of upper surface of concrete backfill which bears directly against the structure (i.e., no seismic isolation material), or

b. Elevation of a continuous floor slab which does not amplify the ground response, whichever is higher.

The actual stiffness of resisting elements that extend below the assumed point-of-fixity is appropriately considered and reflected in the calculation of pseudo beam properties.

Equipment having relatively small mass/or high frequency is decoupled from the supporting structure, but its mass is included with the mass of the supporting system. The NSSS, a major equipment system whose dynamic behavior can have important interaction with the supporting containment concrete internals structure, was modeled as a coupled system, NSSS and structure, by Westinghouse in their dynamic analyses. The structural portion of the mode is represented by Figure 3.7(B)-22.

The hydrodynamic mass effects of the fluid inside the structure are considered in modeling the inertia properties. Equivalent weights of fluid that are effective in producing impulsive forces and the constrained weight of fluid are determined and included in the lumped mass model at the appropriate nodes. These weights were calculated using the method described in Reference 1. The combined effects of the two horizontal and one vertical motion are taken into account to obtain the design parameters. The three seismic responses or effects at a particular point caused by each of the three orthogonal components of seismic motion are combined by taking the square-root-of-the-sum-of-the-squares of the particular effect or response at that point, in accordance with Regulatory Guide 1.92. For symmetric structures, such as the containment, coupling between translational and torsional modes of vibration is negligible, and the dynamic degrees of freedom are therefore uncoupled for the two horizontal directions. The seismic analysis of structures in the vertical direction is performed separately considering different mathematical models of the structures. For this analysis, only the vertical dynamic degrees-of-freedom associated with each mass are retained.

Ground response spectra for the Seabrook site (Subsection 3.7(B).1) are in accordance with Regulatory Guide 1.60, factored by 0.25 to match the site maximum acceleration. These spectra are valid for structure and component design provided the system in question is supported on bedrock or concrete fill over bedrock, or is a portion of a structure which is subjected to ground response. The ground response spectra or artificial time-histories are applied as input at the structure's point-of-fixity. The lumped-mass models of Figure 3.7(B)-21, Figure 3.7(B)-22, Figure 3.7(B)-23, Figure 3.7(B)-24 and Figure 3.7(B)-25 have the properties listed in Table 3.7(B)-2, Table 3.7(B)-3, Table 3.7(B)-4, Table 3.7(B)-5, Table 3.7(B)-6, Table 3.7(B)-7 and Figure 3.7(B)-8, as noted previously in Subsection 3.7(B)2.1.

3.7(B).2.4 Soil/Structure Interaction

All major seismic Category I structures are founded on rock or concrete extending to rock, thus permitting a fixed base approach to be used. Static and dynamic earth pressures for Category I structures surrounded by offsite borrow are discussed in Subsection 2.5.4.10c, and the pressure diagrams are shown on Figure 2.5-66 and Figure 2.5-67. Analysis methods for computing static and dynamic earth pressures on seismic Category I structures are provided in Subsection 2.5.4.10c.

There are no hydrodynamic effects due to groundwater because there are no structures exposed to free groundwater. The only dynamic effect produced by the presence of groundwater is an increase in dynamic lateral soil pressure resulting from an increase in density of backfill material from that of the moist condition to that of the saturated condition.

Certain seismic Category I electrical manholes are founded on a thin stratum of soil (maximum depth of soil between the foundation and the bedrock is 18 feet), as described in Subsection 3.7(B).1.4. These manholes are analyzed using the multiple lumped mass-spring approach described by Whitman in Reference 2.

All seismic Category I manholes are completely embedded. The effect of embedment is to increase the soil spring stiffness thus increasing the natural frequency of the system resulting in reduced seismic design value. Hence, it was conservatively assumed to neglect embedment effect. The input design ground response spectra, discussed in Subsection 3.7(B).1.1, which were used for the analysis indicated that the increase in frequency would result in decrease in response.

To reduce amplification properties of the soil between the ground surface and the rock, the backfill material is controlled by controlling the placement requirement and material characteristics. The soil property given in Updated FSAR Subsection 3.7(B).1.4 is used for determining soil stiffness. The variation in soil parameters was not considered for the analysis. The lowest shear wave velocity was used to obtain lowest structural frequency which would give higher structural responses as stated above.

The above conservative assumption combined with lower system damping values of 7 percent for SSE and 4 percent for OBE causes seismic load in excess of the actual value.

The design of manholes is governed by dynamic soil pressure; the inertia loads had a minimum effect on the design.

3.7(B).2.5 Development of Floor Response Spectra

The analysis for overall dynamic response of the structural system is described in Subsections 3.7(B).2.1 through 3.7(B)2.3. Local amplification of overall response (generally of floor slabs in the vertical direction) is represented by amplified response spectra (ARS). A time-history seismic analysis is used to generate these spectra using either of two methods. In the first method, the local amplification of slabs, beam and columns is evaluated and an appropriate range of frequencies selected for all local frequencies below 33 Hz. Single-degree-of-freedom (SDOF)' systems representing the computed range of local frequencies are connected to the stick model of the overall structure such that the total weight at the elevation in question (i.e., summation of the weight of the SDOF nodes plus the stick model node) equals the total weight, as if the slabs, beams, columns and walls were all rigid. The total stick model is then analyzed using the ground motion artificial time histories described in Subsection 3.7(B).1 as the input forcing function. The second method of evaluation of local amplification is to model the sub-structure with finite elements in sufficient detail to predict local modes of vibration. The response time-history from the overall stick model, at the elevation of the sub-structure, is then used as the input forcing function.

For symmetric structures, separate analyses are performed for two horizontal directions, and the individual time-history motions and amplified response spectra are obtained for the nodal point locations of the lumped-mass model. For unsymmetric structures, the floor response spectra values in both horizontal directions are initially obtained due to each horizontal component of motion. The resultant response spectra for each horizontal direction are then obtained by combining co-directional responses according to the square-root-of-the-sum-of-the-squares method. The floor response spectra in the vertical direction are obtained separately from the vertical response time-history motions. No coupling of vertical and horizontal motion exists, since rock foundation precludes rocking motion.

Structural damping used in the seismic analysis of all Seabrook structures is in conformance with Regulatory Guide 1.61. Amplified response spectra are generated at 1, 2 and 4 percent equipment damping associated with the OBE, and at 1, 2, 3, 4 and 7 percent equipment damping for the SSE.

To obtain complete and accurate response spectra with respect to peak values, the spectrum ordinates are calculated at natural frequencies of supporting structures where peaks are normally expected, and at other frequencies at sufficiently small frequency intervals. Frequencies listed in Table 3.7(B)-21 are used to compute floor response spectra. This table is based on Table N-1226-1 of the ASME Boiler and Pressure Vessel Code, Section III, Division I, Nuclear Power Plant Components, 1980 Edition, Appendix N, 'Dynamic Analysis Methods.' The floor response spectra generated based on Table 3.7(B)-21 including the natural frequencies of supporting structures will meet the intent of SRP Subsection 3.7.1 (and also R.G. 1.122). The response pectra developed at the subsystem supports as described above, and widened and enveloped as described in Subsection 3.7(B).2.9, are used directly, for the analysis of the subsystem. See Figure 3.7(B)-26, Figure 3.7(B)-27, Figure 3.7(B)-28 and Figure 3.7(B)-29 for typical amplified response spectra for representative seismic Category I structures. Figure 3.7(B)-35 and Figure 3.7(B)-36 show typical comparisons of enveloped floor response spectra for 1 and 4 percent damping generated using frequency intervals based on Table 3.7(B)-21 and R.G. 1.122. A single absolute maximum spectrum is generated from the appropriate response points if more than one set of spectra are generated for a particular elevation or portion of a floor area.

3.7(B).2.6 Three Components of Earthquake Motion

The seismic analyses of the Category I structures are performed using three-dimensional lumped mass models of the structures. The maximum response in principal directions is calculated using each of the three components of earthquake motion, and then the responses in each direction are obtained by taking the square-root-of-the-sum-of-the-squares of the maximum co-directional responses at a particular point in the structure. This procedure is in accordance with Standard Review Plan 3.7.2, Section II.6a and b(1).

3.7(B).2.7 Combination of Modal Responses

Modal responses are combined as per Regulatory Guide 1.92, paragraph 1.1 and 1.2.1. When the response spectrum method of analysis is used to determine seismic response of seismic Category I structures, the most probable response is obtained as the square-root-of-the-sum-of-the-squares of the responses from the individual modes.

For the closely spaced modes having frequencies within 10 percent of each other, the structural response is obtained by first obtaining the absolute sum of the responses of the closely spaced modes and then combining this sum with other remaining modal responses using the square-root-of-the-sum-of-the-squares method.

See Subsection 9.1.4.3.a.5(d)(1) for the methodology used in support of upgrading the Cask Handling Crane to meet the single failure proof requirements of NUREG-0554.

3.7(B).2.8 Interaction of Non-Category I Structures with Category I Structures

All nonseismic Category I structures which, due to their proximity to seismic Category I structures, could possibly compromise the safety function of the seismic Category I structures by their collapse, are either designed for SSE loading or are designed to collapse away from the adjacent seismic Category I structures (see Table 3.7(B)-22 for further discussion). The methods described in Subsection 3.7(B).2.1 are used for the analyses of these structures.

3.7(B).2.9 Effects of Parameter Variation on Floor Response Spectra

The location of the peak responses on amplified response spectra curves vary as a result of the variation in material properties. This impacts subsystem design as discussed in Subsection 3.7(B).3. The variability in structural materials' properties and modeling assumptions is accounted for by peak spreading when generating envelopes of the response spectra. All Category I structures for which in-structure response spectra are generated are supported on rock and, hence, variability of soil properties is not the consideration in broadening the peaks of floor response spectra. A value of at least ± 10 percent is used to broaden the peaks of the floor response spectra associated with the structural frequencies. Regulatory Guide 1.122 is complied with except as noted in Section 1.8.

3.7(B).2.10 Use of Constant Vertical Load Factors

Detailed vertical seismic system analyses are performed for seismic Category I structures; therefore, constant vertical static factors are not used.

3.7(B).2.11 Method Used to Account for Torsional Effects

Torsional degrees-of-freedom are included in the dynamic analysis to obtain the natural frequencies and mode shapes of seismic Category I structures. For structures in which the center of mass and center of rigidity is not coincident, torsional effects are automatically considered when the response to horizontal motion is obtained. The design story shears are computed by considering both the direct shear from the orthogonal earthquake components and the shear due to torque.

An accidental torsion, based on 5 percent eccentricity (SRP 3.7.2, Rev. 1, July 1981) was shown to have negligible effect on the design.

3.7(B).2.12 Comparison of Responses

The nodal displacement and acceleration responses obtained from response spectrum and time-history methods of analysis are compared for the Containment Building and Fuel Storage Building. Table 3.7(B)-10 and Table 3.7(B)-11 list the displacements and accelerations computed by both methods for horizontal and vertical directions and SSE and OBE conditions (as noted on the tables) for the Containment Building. Referenced elevations correspond to mass points as shown on Figure 3.7(B)-21. Table 3.7(B)-19 shows similar information for the Fuel Storage Building, whose model is shown on Figure 3.7(B)-25. Approximate equivalency can be seen in the results of the two methods.

3.7(B).2.13 Methods for Seismic Analysis of Category I Dams

This section is not applicable to Seabrook.

3.7(B).2.14 Determination of Seismic Category I Structure Overturning Moments

All seismic Category I structures are designed to resist overturning due to the combined effects of the vertical and two horizontal components of seismic ground motion. A structure's ability to resist overturning is calculated by either of two conservative approaches: moment equilibrium or a work-kinetic energy method.

a. Moment Equilibrium Method

In the moment equilibrium method, the response of a structure due to three directions of earthquake is obtained from the dynamic analyses. The maximum overturning moment, about the toe of the mat, is computed as:

$$M_O = M_{h1} \text{ or } M_{h2}$$

where: M_O = overturning moment

M_{h1} = maximum overturning moment in one horizontal direction due to effects of three directions of earthquake.

M_{h2} = maximum overturning moment in second horizontal direction due to effects of three directions of earthquake.

M_{h1} or M_{h2} include all dynamic effects on the structure including the dynamic effect of soil caused by seismic motions.

The resisting moment is computed as follows:

$$M_R = (W - V_S)\,X_1 - (V_h)\,X_2 + M_b$$

where: M_R = resisting moment

W = weight of the structure and any fill

V_s = maximum vertical seismic force of structure acting upward due to three directions of earthquake

V_h = maximum vertical hydrostatic force on the structure acting upward

X_1 = horizontal distance of the centroid of the structure from the toe of the mat

X_2 = horizontal distance to center of hydrostatic area from the toe of the mat

M_b = resisting moment due to key action of mat or passive resistance of the structure.

The factor of safety against overturning is:

$$FS = \frac{M_R}{M_O}$$

As long as the factor of safety is equal to or greater than 1.10 for SSE and 1.50 for OBE load conditions, the structure is considered stable against overturning.

b. Work-Kinetic Energy Method

In the work-kinetic energy method, the kinetic energy in a structure due to an OBE or SSE is estimated by:

$$KE = \frac{1}{2}\sum M_i \left[(V_H)_i^{\,2} + (V_V)_i^{\,2}\right]$$

M_i = Mass concentration at some point (i) in the structure

$(V_H)_i$ = Maximum total lateral velocity

$(V_v)_i$ = Maximum total vertical velocity

$(V_H)_i$ and $(V_v)_i$ are computed as follows:

$(V_H)_i^2 = (V_x)_i^2 + (V_H)_g^2$

$(V_v)_i^2 = (V_z)_i^2 + (V_v)_g^2$

where:

$(V_H)_g$ = Peak horizontal ground velocity

$(V_v)_g$ = Peak vertical ground velocity

$(V_x)_i$ = Maximum relative lateral velocity of mass, M_i, due to three directions of earthquake

$(V_z)_i$ = Maximum relative vertical velocity of mass, M_i, due to three directions of earthquake.

$(V_H)_g$ and $(V_v)_g$ are obtained from ground response spectra of the time history used in seismic analysis. $(V_x)_i$ and $(V_z)_i$ are obtained from seismic analysis of the structure.

The work, W, required to overturn the structure is computed as:

$$W = M_t\, g\, h + W_p - W_b$$

where:

M_t = Total mass of the structure and foundation mat

g = Gravitational acceleration

h = The vertical distance for which the center of mass of a structure must be lifted to reach the overturning position

W_p = The additional work required to displace the soil on the toe side of an embedded structure

W_b = The work done by the buoyancy force on the submerged portion of a structure.

Each collinear effect (as described in both a. and b. above) due to the effects of three directions of earthquake is obtained by combining the respective individual components using the SRSS method.

The structure is considered stable against overturning when the ratio W/KE exceeds the safety factors described in a. above.

3.7(B).2.15 Analysis Procedure for Damping

When the components of a seismic Category I structure or system are constructed of different materials, and these components cannot be uncoupled due to dynamic interaction effects, an average modal damping value is used for the dynamic analysis of the system. This average modal value is computed from the damping values of the various components, as listed in Table 3.7(B)-1, each weighted by the energy stored in these components in the various modes of vibration. This value is computed as:

$$D_n = \frac{\sum D_i E_{in}}{E_{Tn}}$$

where:

D_n	=	Average damping value of the n^{th} mode
D_i	=	Damping value of the i^{th} component
E_{in}	=	Energy stored in the i^{th} component in the n^{th} mode
E_{Tn}	=	Total energy stored in the structure or system in the n^{th} mode.

The energy values are computed from the stiffness matrices of the various components and the mode shapes associated with undamped vibration.

By this procedure, a diagonal damping matrix is computed allowing for the uncoupling of the equations of motion and the use of modal super-position.

3.7(B).3 Seismic Subsystem Analysis

This section describes the seismic analysis performed on all subsystems, exclusive of those within the NSSS supplier's scope of responsibility. See Subsection 3.7(N).3 for the NSSS supplier's discussion.

3.7(B).3.1 Seismic Analysis Methods

The seismic analyses of all Category I subsystems and components use either a dynamic analysis method or an equivalent static load method. Selection of the particular method of analysis depends upon the suitability of the mathematical model to adequately represent the behavior of the system or component. All points of the systems or components where the deflections, loads and stresses are expected to be significant are included in the mathematical model. The use of the equivalent static load method is limited to systems which can be represented by simple mathematical models.

a. Dynamic Analysis

Dynamic analyses were performed on subsystems and components such as electrical cable tray supports, electrical conduit supports, instrument racks, battery racks, etc., using the modal response spectrum analysis technique. The mathematical model for the electrical cable tray supports, as represented by Figure 3.7(B)-30, is typically represented by Figure 3.7(B)-31. Modeling considerations are discussed in Subsection 3.7(B).3.3b.

The number of modes considered in determining the response of the equipment or components is such that inclusion of additional modes would not result in more than a 10 percent increase in the response.

b. Equivalent Static Analysis

The equivalent static analysis method consists of applying a load at the center-of-gravity of the component or equipment in the direction of seismic excitation, where the applied load is calculated by multiplying the total weight of the component or equipment with the applicable seismic acceleration level corresponding to the fundamental frequency of the component or equipment, using an appropriate static coefficient to account for the combined modal participation of the higher modes of vibration. When the equipment or component fundamental frequency is less than 33 Hertz, a static coefficient of 1.5 will be used, except where a reduced static coefficient can be justified. When the equipment or component fundamental frequency is equal to or greater than 33 Hertz, a static coefficient of 1.0 will be used when the equipment or component has a single predominant frequency. When there is more than one predominant frequency, a static coefficient of 1.3 will be used, except where a reduced static coefficient can be justified. The criteria for selecting the applicable seismic acceleration level are as follows:

1. When the equipment or component fundamental frequency is greater than 33 Hertz, the acceleration level corresponds to the 'g' value of the rigid range of the amplified floor response spectra.

2. When the equipment or component fundamental frequency is less than or equal to 33 Hertz, or is not evaluated, the acceleration level corresponds to the peak 'g' value of the amplified floor response spectra. However, if the equipment or component fundamental frequency is predominant over all other frequencies, the actual 'g' value corresponding to the fundamental frequency is used, except that such 'g' value must be equal to or greater than the maximum 'g' values corresponding to all higher frequencies.

3.7(B).3.2 Determination of Number of Earthquake Cycles

During the forty-year design life of the plant, the occurrence of one safe shutdown earthquake (SSE) and five operating basis earthquakes (OBE) are postulated. For fatigue analysis requirements of safety Class 1 components, a maximum of twenty stress cycles per earthquake is postulated, which results in a total of one hundred cycles for the five OBE events. For the SSE, no estimate of the number of stress cycles is made, since it is a faulted condition, and a fatigue analysis is not required.

3.7(B).3.3 Procedure Used for Modeling

a. Modeling of Piping Systems for Dynamic Analysis

The piping systems were analyzed using a three-dimensional structural model composed of concentrated lumped masses connected by massless spring elements having the same strength and stiffness properties as the pipe. The model accounts for the interaction effect between piping, equipment and supports. Supports were modeled as flexible members with the appropriate spring rate to represent the support stiffness. The piping model was terminated at equipment nozzles that were modeled as rigid anchors with consideration given to the seismic amplification of equipment, as follows:

1. For rigid equipment in which the fundamental frequency was equal or higher than 33 Hz, the amplified response spectra of the structure was used.

2. For equipment in which the fundamental frequency was lower than 33 Hz, the amplified response spectra and the seismic anchor displacement of the equipment at the pipe/nozzle interface point was used.

All in-line components were included in the model. The concentrated mass of in-line components such as valves, flanges, and strainers were represented as lumped masses. Valve operators were modeled as an offset lumped mass to account for the torsional and in-plane bending effects on the piping.

The following criteria were used for the decoupling of piping subsystems:

1. Piping was decoupled from the equipment, and the nozzle modeled as a full, six-degree-of-freedom restraint.

2. Branch connections were decoupled from the main runs when the ratio of the branch to run section moduli was equal to or less than 0.05.

3. Piping subsystems which were decoupled into separate analytical models satisfied one of the following criteria:

   (a) The boundary of the decoupling point is a full anchor for the piping of both separate models.

   (b) The boundary of each decoupled model contains a region of common overlap to both models which provides restraint(s) in each of the three orthogonal directions.

b. <u>Modeling of Equipment</u>

Seismic Category I equipment is modeled as lumped-mass spring systems which consist of a series of discrete mass points connected by massless elastic members. All significant concentrated weights are represented as lumped masses. Typical examples of concentrated weights are weights of motor rotor and pump impeller in the analysis of shafts. In general, masses are lumped at points where the maximum displacements are anticipated to occur. The number of lumped masses are such that the mathematical model represents the equipment response as closely as possible.

3.7(B).3.4 Basis for Selection of Frequencies

All frequencies below 33 Hz are considered in computing the total response of the equipment and components. Whenever possible, equipment and components are designed so that their fundamental frequencies are less than half or more than twice the dominant frequencies of the support structure. Where this is found to be impractical or impossible to achieve, the equipment and components are adequately designed for the amplified loading.

3.7(B).3.5 Use of Equivalent Static Load Method of Analysis

The criteria and procedure for the use of the equivalent static load method are described in Subsection 3.7(B).3.1b.

3.7(B).3.6 Three Components of Earthquake Motion

Seismic Category I subsystems and components are analyzed by considering the combined effects of seismic loads occurring in three mutually perpendicular directions, two in the horizontal direction and one in the vertical direction. The total combined response (displacements, stresses, and forces) due to the three components of earthquake motion is obtained by using the square root of the sum of the squares method as follows:

$$X_c = \sqrt{\sum_{j=1}^{3} X_j^{\,2}}$$

Where X_c = Total combined response of the parameter x (displacement, stress or force etc.)

X_j = Value of the combined response of the parameter x in the j-direction of the earthquake.

3.7(B).3.7 Procedures for Combining Modal Responses

The combined response of equipment and components in a given earthquake direction is obtained by combining the individual modal responses using the square-root-of-the-sum-of-the-squares method. For equipment and components having modes with closely spaced frequencies, this method is modified to include the possible effects of these modes. The groups of closely spaced modes are chosen such that the difference between the frequencies of the consecutive modes in a group does not exceed ten percent. No one frequency is in more than one group. For closely spaced modes, the modal response in a particular earthquake direction is obtained by using Equation (1) for piping and associated in-line components and by Equation (2) for equipment and other components.

For closely spaced modes, X_j is given by:

$$X_j = \left[\sum_{k=1}^{N'} X_k^{\,2} + \sum_{q=1}^{P} \left\{ \sum_{m=1}^{nq} \left| X_{mq} \right| \right\}^2 \right]^{1/2}$$

Where, $N' = N - \sum_{q=1}^{P} n_q$

P	=	Number of groups of closely spaced modes
n_q	=	Number of closely spaced modes in group number q
N	-	Total number of modes
X_{mq}	=	Maximum value of the response of the element attributed to the mth mode of group number q
X_K	=	Maximum value of the response of the element due to Kth mode

An alternate expression for X_j is given by:

$$X_j = \left[\sum_{k=1}^{N'} X_k^{\ 2}\right]^{1/2} + \sum_{m=1}^{R} \left|X_m\right|$$

Where the variables are as defined above except for the following:

N'	=	N-R
R	=	Modes with closely spaced frequencies.
X_m	=	Maximum value of the response of the element attributed to the mth mode.

The combined response of piping in a given earthquake direction is obtained by combining the individual modal responses using the square-root-of-the-sum-of-the-squares (SRSS) method. The closely spaced mode technique described in Subsection 3.7(B).3.7 may also be used.

3.7(B).3.8 Analytical Procedures for Piping

The procedure used for modeling of piping systems is described in Subsection 3.7(B).3.3a. The analytical procedures applicable to piping systems are as follows:

a. Dynamic Analysis of Piping Systems

Detailed seismic analysis of piping systems is performed by using finite element analysis programs to ensure that the stresses in the piping system meet the applicable ASME Section III Code requirements. Modal Response Spectrum Analysis technique is used for the dynamic analysis of piping systems. The seismic loading input consists of amplified floor response spectra obtained for discrete structural locations.

The piping system is represented by a three-dimensional lumped-mass model, and is analyzed by determining the response of the system to the three components of the earthquake motion. The procedure for determining the total combined response is described in Subsection 3.7(B).3.6.

b. Procedures to Account for Differential Piping Support Movement

The piping system is analyzed for differential support movement at different support points located within a building and between buildings by using the static analysis method, the maximum relative support displacements are obtained from structural response calculations, and the worst differential movements between support points is used for the piping analysis.

Pipe Stresses from support movement analysis are combined with the other stresses in appropriate Code equations. Loads on supports are added to other applicable loads.

c. Equivalent Static Analysis of Piping Systems

When seismic analysis of piping systems is performed using the static analysis approach, the total combined response (displacements, stresses, and forces) due to the three components of earthquake motion is obtained by using the square-root-of-the-sum-of-the-squares method. The response of the above parameters in each earthquake direction is obtained by applying a uniformly distributed 'g' load in that direction multiplied by a static coefficient of 1.5. The magnitude of the 'g' value corresponds to the fundamental frequency of the system, and is obtained from the applicable amplified response spectra curve.

3.7(B).3.9 Multiply-Supported Equipment Components with Distinct Inputs

When the response spectrum method is used to analyze equipment and components supported at different elevations within a building or between buildings, typically, an upper bound envelope of all the individual response spectra for these locations is used to calculate the inertial responses of the multiply-supported equipment and components. In addition, the relative displacements at the support points are determined according to the procedure outlined in Subsection 3.7(B).3.8b.

Where the response of selected items in the analytical model is under the direct influence of a specific support point amplified response spectra, then this analysis may be performed using this amplified response spectra. Additionally, where necessary, alternate methods are used which employ acceptable techniques which consider multiple amplified response spectra.

3.7(B).3.10 Use of Constant Vertical Static Factors

Constant vertical static factors are not used for the design of seismic Category I subsystems.

3.7(B).3.11 Torsional Effect of Eccentric Masses

The mathematical model generated to represent a piping system includes consideration of eccentric masses, such as valve operators.

The eccentric mass is represented in the model as a lumped mass at the end of a weightless beam cantilevered from the valves. The length of the beam is equal to the distance between the center of gravity of the operator and the axis of the pipe.

3.7(B).3.12 Buried Seismic Category I Piping Systems and Tunnels

Seismic Category I, ASME Section III, piping systems have been analyzed by utilizing the schemes and procedures developed by Newmark and Rosenblueth (Reference 3), Shah and Chu (Reference 4) and Iqbal and Goodling (Reference 5).

The criteria for analysis of buried piping systems is to demonstrate their capability to withstand seismic soil strain and internal pressure loads without exceeding code allowable stress levels. Analysis of these underground pipes subjected to seismic ground motion is based on the piping system configuration, boundary conditions, and the elastic properties of the soil and piping.

Seismic Category I, ASME Section III, buried piping which penetrates structures where seismic soil-structure differential movements are expected is protected by providing adequate clearance between pipe and sleeve at the penetrations or by additional bend for flexibility.

There are no seismic Category tunnels which are analyzed as subsystems. Electrical, piping and passage tunnels are analyzed as seismic Category I structures, and are discussed in Subsection 3.7.2.

3.7(B).3.13 Interaction of Other Piping with Seismic Category I Piping

The loading effects of nonseismic Category piping portions on seismic Category I piping is kept to a minimum by providing a restraint, or a group of restraints, at or beyond the defined boundaries, and extending the seismic analysis performed for the seismic Category I portion of the piping system to the boundary restraints. These restraints are designed to withstand the most severe loading combinations that could result from the nonseismic Category I piping.

More specifically, to assure that an earthquake of SSE intensity will not result in failure of seismic Category I piping, the mathematical models constructed for Category I piping include the nonseismic Category I piping up to a boundary restraint (defined as a set of one or more restraints or anchor). Typical functions and locations of these restraints are shown in Figure 3.7(B)-37. In Case 1, moments and forces introduced by the nonseismic piping are reacted through the six degree-of-freedom restraint as shown. In Case 2, moments are reacted by couples between pairs of restraints which limit motion in the indicated directions. These boundary restraints are designed to withstand the plastic capability of the contiguous piping.

The structural integrity of the nonseismic Category I piping is assured by compliance with the stress limits shown on Figure 3.7(B)-37. Class 2 or 3 piping must satisfy the limits of ASME III, NC-3600 or ND-3600. Pipes and supports in the shaded regions are included in the mathematical model of the seismic Category I piping and are subjected to SSE loads. Nonseismic piping beyond the boundary restraints satisfies the limits of B31.1. Loading conditions and stress limits for all pipe classifications are summarized in Subsection 3.9.3.

3.7(B).3.14 Seismic Analysis of Reactor Internals

See Subsection 3.7(N).3.14.

3.7(B).3.15 Analysis Procedure for Damping

Where the equipment or component consists of subcomponents with the same damping characteristics, the same critical damping value is used for the entire equipment or component. The corresponding critical damping value is chosen from Table 3.7(B)-23. For seismic Category I equipment or component consisting of subcomponents with different damping characteristics, the lowest critical damping value associated with the subcomponents in the equipment or component is used in the analysis for all modes.

Alternate critical damping values are used for a specific component or subsystem, where documented test data justifies such usage.

3.7(B).4 Seismic Instrumentation

In the event of an earthquake, the seismic instrumentation will provide information on the input ground motion and resultant vibratory responses of representative Category I structures and equipment so that an evaluation can be made as to:

- Whether input design response spectra were exceeded
- Whether the vibratory responses of the representative Category I structures and equipment were exceeded
- The need for shutdown of the plant
- The degree of validity of the mathematical models used in the seismic analysis of the buildings and equipment.

The design requirement of the seismic instrumentation is based on a Safe Shutdown Earthquake (SSE) of 0.25g (peak) and time duration of fifteen (15) seconds, and an Operating Basis Earthquake (OBE) of 0.125g (peak).

The SM instrumentation is not Class 1E qualified. It is not designed for a design basis event environment, except the SSE. It is ANS Safety Class NNS and Seismic Category I.

3.7(B).4.1 Comparison with Regulatory Guide 1.12

The SM system instrumentation type, number and locations described follow the guidance provided in USNRC Regulatory Guide 1.12, revision 1 (RG 1.12), 1974, with exceptions as discussed below. These exceptions are consistent with draft Regulatory Guide DG-1016, "Nuclear Power Plant Instrumentation for Earthquakes."

Triaxial peak accelerographs, recommended by RG 1.12, for attachment to reactor equipment and piping inside containment, and to Seismic Category I equipment or piping outside containment, are not installed as part of the SM system.

All locations where response spectrum recorders are recommended by Regulatory Guide 1.12 are monitored by triaxial time history accelerometers. Triaxial time history accelerometer XT-6701, linked to the Condor (Kinematics product name) digital recording system, provides the seismic response of the Containment foundation.

RG 1.12 recommends the installation of accelerometers on two elevations (not including the foundation) of the Containment building. Triaxial time history accelerographs are installed on the Containment foundation at elevation (-)26', and the operating deck at (+)25'.

RG 1.12 recommends the installation of accelerometers on two independent Seismic Category I foundations (not including the Containment foundation). One triaxial time history accelerograph is installed in the Service Water Pump House at elevation 22'.

One triaxial time history accelerograph is installed on elevation 53' in the PAB, not on two elevations of two independent Seismic Category I buildings as recommended by RG 1.12.

3.7(B).4.2 Location and Description of Instrumentation

The SM instrumentation described below is intended to provide timely and accurate technical information necessary for an informed assessment of the integrity of safety-related components, systems and structures immediately following an earthquake.

Electric power for seismic instrumentation is provided from a Class 1E, 120V, uninterruptable power supply.

a. Time History Accelerograph System Condor

Seismic instrumentation is provided, one each at the following locations:

1. At a free field position in the control room east air intake, on bedrock.

2. Between columns 16 and 17 in the containment building foundation at elevation (-) 26'-0".

3. Between columns 16 and 17 on the concrete operating floor in the Containment Building at elevation (+)25'-0".

The triaxial accelerometer (type FBA-3) has a frequency range from 0.0 to 50.0 Hz. Each triaxial accelerometer is connected to a digital recorder that is part of the Condor system located in seismic control panel 1-SM-CP-58. The trigger method is a threshold exceedance type. Its threshold level is set at 0.01g to avoid actuation due to insignificant motion but to record a seismic disturbance which creates significantly lower ground accelerations than that of the OBE (0.125g).

The two seismic instrumentation packages located in the Containment building are oriented so that the axes of the accelerometers are pointing in the same direction and are aligned to the axis of the building.

The Kinematics solid state Condor system is a central recording, time history accelerograph and display system. The recorder's digital triggering system continuously monitors the signals sent by remote accelerometer. When motion exceeds the preprogrammed threshold, the recorders begin sampling concurrently on all channels. The data is automatically recorded with information such as sensor location, event time and sensor serial number. When the system detriggers, the central controller interrogates the recorders and downloads the event files into a dedicated PC for automatic OBE analysis. The total capacity of the event recording is at least 25 minutes.

b. Solid State Accelerographs (ETNA, Kinemetrics product name)

Two stand-alone solid state accelerographs, ETNA, are installed in the following locations:

1. In the Primary Auxiliary Building, PAB, at elevation 53'.

2. On the southwest corner of the electrical control room of the Service Water (SW) Pumphouse at elevation 22'-0".

Each recorder can trigger and record a seismic event using a preset threshold level as described for the Condor system. Each ETNA recorder has an internal triaxial accelerometer with a full scale of ±2g, a frequency response from 0.0 to 50.0 Hz, and a 70% damping coefficient.

c. Criteria for Instrument Location

The selection of the above locations is based on the guidance provided in USNRC Regulatory Guide 1.12, Revision 1, for an SSE acceleration of less than 0.3g with exceptions as provided in Subsection 3.7(B).4.1.

All instruments are accessible for inspection, test and service.

Table 3.7(B)-24 summarizes the location of each primary instrument.

3.7(B).4.3 Control Room Operator Notifications

Audio visual alarms are provided on the main control board for the following parameters:

a. "SEISMIC EVENT IN PROGRESS"

b. "SEISMIC MONITOR TROUBLE"

c. "OBE EXCEEDANCE" (only for east air intake)

These conditions apply to instruments at the following locations:

1. East air intake
2. Containment building foundation
3. Containment operating floor

3.7(B).4.4 Comparison of Measured and Predicted Responses

The Condor is an automatic data retrieval and analysis system based on a high-speed, rack-mounted computer (Central Controller). The system remains on at all times monitoring the recorder triggering activity. When an event has been recorded, the Central Controller retrieves the time history data and automatically evaluates OBE exceedance criteria including a computation of response spectra and cumulative absolute velocity (CAV). A printer allows immediate print and plot results for operator review.

Detailed comparisons are made between SM system measured responses and calculated responses based on plant dynamic models.

The time history records from the east air intake are used to calculate response spectra at the appropriate critical damping ratio. This is compared to the design response spectra.

The response spectra measured at the Service Water Pumphouse foundation and the Primary Auxiliary Building at elevation 53' are compared with those calculated by using the time history records from the east air intake as input ground motion to the containment and primary auxiliary building dynamic models. These comparisons indicate the validity of the dynamic model and form the basis for adjustment of the model.

Measured structural responses and response spectra are compared against the original design and analysis parameters to permit evaluation of the seismic effect on structures and equipment. These comparisons provide the basis for a detailed physical inspection of structures and equipment.

3.7(B).5 In-Service Surveillance

Calibration and alignment on three orthoginal axes are performed prior to fuel loading to assure proper operation. Periodic testing and calibration is performed in accordance with Technical Requirements.

3.7(B).6 References

1. "Nuclear Reactors and Earthquakes," TID-7024, Prepared by Lockheed Aircraft Corporation and Holmes & Narver, Inc., for the Division of Reactor Development, U. S. Atomic Energy Commission, Washington, D. C., August 1963.

2. Whitman, R.V., 1969: "Equivalent Lumped System for Structure Founded Upon Stratum of Soil," Proc. 4th World Conf. on Earthquake Engineering, Chile, Vol. III, Section A-6, pp. 133-142.

3. Newmark, N. M. and Rosenblueth, E., "Fundamentals of Earthquake Engineering," Prentice-Hall, 1971.

4. Shah, H. H. and Chu, S. L., "Seismic Analysis of Underground Structural Elements," Journal of the Power Division, Proceedings of the American Society of Civil Engineers, Vol. 100, No. P01, July 1974, PP. 53-62.

5. Iqbal, M. A. and Goodling, E. C., "Seismic Design of Buried Piping," and ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities, December 1975.

3.7(N) SEISMIC DESIGN

In addition to the steady-state loads imposed on the system under normal operating conditions, the design of equipment and equipment supports requires that consideration also be given to abnormal loading conditions such as earthquakes. Seismic loadings are considered for earthquakes of two magnitudes: Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE). The SSE is defined as the maximum vibratory ground motion at the plant site that can reasonably be predicted from geologic and seismic evidence. The OBE is that earthquake which, considering the local geology and seismology, can be reasonably expected to occur during the plant life.

For the OBE loading condition, the Nuclear Steam Supply System is designed to be capable of continued safe operation. The design for the SSE is intended to assure:

a. That the integrity of the reactor coolant pressure boundary is not compromised

b. That the capability to shutdown the reactor and maintain it in a safe condition is not compromised

c. That 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 is not compromised.

The seismic qualification requirements for safety-related instrumentation and electrical equipment are covered in Section 3.10. The seismic and ANS safety class definitions and classification lists are given in Section 3.2. Operating condition categories and the methods used for seismic qualification of mechanical equipment are given in Section 3.9.

3.7(N).1 Seismic Input

3.7(N).1.1 Design Response Spectra

Refer to Subsection 3.7(B).1.1.

3.7(N).1.2 Design Time History

Refer to Subsection 3.7(B).1.2.

3.7(N).1.3 Critical Damping Values

The damping values given in Table 3.7(N)-1 were used for the systems analysis of Westinghouse equipment. These are consistent with the damping values recommended in Regulatory Guide 1.61, except in the case of the primary coolant loop system components and large piping (excluding reactor pressure vessel internals) for which the damping values of 2 percent and 4 percent are used, as established in testing programs reported in Reference 1. (The damping values of ASME B&PV Code, Code Case N-411, may be used for pipe stress verification and for pipe support optimization in place of the foregoing). The damping values for control rod drive mechanisms (CRDM) and the fuel assemblies of the Nuclear Steam Supply System, when used in seismic system analysis, are in conformance with the values for welded and/or bolted steel structures (as appropriate) listed in Regulatory Guide 1.61.

Tests on fuel assembly bundles justified conservative component damping values. Documentation of the fuel assembly tests is found in Reference 7.

The damping values used in component analysis of CRDM and their seismic supports were developed by testing programs performed by Westinghouse. These tests were performed during the design of the CRDM support; the support was designed so that the damping in Table 3.7(N)-1 could be conservatively used in the seismic analysis. The CRDM support system is designed with plates at the top of the mechanism and gaps between mechanisms. These are encircled by a box section frame which is attached by tie-rods to the refueling cavity wall. The test conducted was on a full-size CRDM complete with rod position indicator coils, attachment to a simulated vessel head, and variable gap between the top of the pressure housing support plate and a rigid bumper representing the support. The internal pressure of the CRDM was 2250 psi and the temperature on the outside of the pressure housing was 400°F.

The program consisted of transient vibration tests in which the CRDM was deflected a specified initial amount and suddenly released. A logarithmic decrement analysis of the decaying transient provides the effective damping of the assembly. The effect on damping of variations in the drive shaft axial position, upper seismic support clearance and initial deflection amplitude was investigated.

The upper support clearance had the largest effect on the CRDM damping with the damping increasing with increasing clearance. With an upper clearance of 0.06 inches, the measured damping was approximately 8 percent. The clearances in a typical upper seismic CRDM support is a minimum of 0.10 inches. The increasing damping with increasing clearances trend from the test results indicated that the damping would be greater than 8 percent for both the OBE and the SSE based on a comparison between typical deflections during these seismic events to the initial deflections of the mechanisms in the test. Component damping values of 5 percent are, therefore, conservative for both OBE and SSE.

These damping values are used and applied to CRDM component analyses by response spectra techniques.

3.7(N).1.4 Supporting Media for Seismic Category I Structures

Refer to Subsection 3.7(B).1.4.

3.7(N).2 Seismic System Analysis

This section describes the methods of seismic analysis performed for safety-related components and systems within Westinghouse's scope.

3.7(N).2.1 Seismic Analysis Methods

Those components and systems that must remain functional in the event of the Safe Shutdown Earthquake (Seismic Category I) are identified by applying the criteria of Subsection 3.2.1.

In general, the dynamic analyses are performed using a modal analysis plus either the response spectrum analysis or integration of the uncoupled modal equations as described in Subsections 3.7(N).2.1c and 3.7(N).2.1d, respectively, or by direct integration of the coupled differential equations of motion described in Subsection 3.7(N).2.1e.

a. Dynamic Analysis - Mathematical Model

The first step in any dynamic analysis is to model the structure or component, i.e., convert the real structure or component into a system of masses, springs, and dash pots suitable for mathematical analysis. The essence of this step is to select a model so that the displacements obtained will be a good representation of the motion of the structure or component. Stated differently, the true inertia forces should not be altered so as to appreciably affect the internal stresses in the structure or component. Some typical modeling techniques are presented in Reference 3.

1. Equations of Motion

Consider the multi-degree of freedom system shown in Figure 3.7(N)-1. Making a force balance on each mass point r, the equations of motion can be written in the form:

$$m_r \ddot{y}_r + \sum^{i} c_{ri} \dot{u}_i + \sum^{i} k_{ri} u_i = 0 \qquad (3.7(N)\text{-}1)$$

m_r = the value of the mass or mass moment of rotational inertia at mass point r

$\ddot{y}_r$ = absolute translational or angular acceleration of mass point r

c_{ri} = damping coefficient - external force or moment required at mass point r to produce a unit translational or angular velocity at mass point i, maintaining zero translational or angular velocity at all other mass points. Force or moment is positive in the direction of positive translational or angular velocity.

u_i = translational or angular velocity of mass point i relative to the base

k_{ri} = stiffness coefficient - the external force (moment) required at mass point r to produce a unit deflection (rotation) at mass point i, maintaining zero displacement (rotation) at all other mass points. Force (moment) is positive in the direction of positive displacement (rotation).

u_i = displacement (rotation) of mass point i relative to the base.

As an example, note that Figure 3.7(N)-1 does not attempt to show all of the springs (and none of the dashpots) which are represented in Equation (3.7(N)-1).

Since:

$$\ddot{y}_r = \ddot{u}_r + \ddot{y}_s \qquad (3.7(N)\text{-}2)$$

Where:

$\ddot{y}_s$ = absolute translational (angular) acceleration of the base

$\ddot{y}_r$ = translational (angular) acceleration of mass point r relative to the base

Equation (3.7(N)-1) can be written as:

$$m_r \ddot{u}_r + \sum^{i} c_{ri} \dot{u}_i + \sum^{i} k_{ri} u_i = m_r \ddot{y}_s \qquad (3.7(N)\text{-}3)$$

For a single degree of freedom system with displacement u, mass m, damping c, and stiffness k, the corresponding equation of motion is:

$$m\ddot{u} + c\dot{u} + ku = -m\ddot{y}_s \qquad (3.7(N)\text{-}4)$$

b. Modal Analysis

1. Natural Frequencies and Mode Shapes

The first step in the modal analysis method is to establish the normal modes, which were determined by eigen solution of Equation (3.7(N)-3). The right hand side and the damping term are set equal to zero for this purpose as illustrated in Reference 4 (pp. 83 through 111). Thus, Equation (3.7(N)-3) becomes:

$$m_r \ddot{u}_r + \sum^{i} k_{ri} u_i = 0 \qquad (3.7(N)\text{-}5)$$

The equation given for each mass point r in Equation (3.7(N)-5) can be written as a system of equations in matrix form as:

$$[M]\{\ddot{\Delta}\} + [K]\{\Delta\} = 0 \qquad (3.7(N)\text{-}6)$$

where:

[M] = mass and rotational inertia matrix

$\{\Delta\}$ = column matrix of the general displacement and rotation at each mass point relative to the base

[K] = square stiffness matrix

$\{\ddot{\Delta}\}$ = column matrix of general translational and angular accelerations at each mass point relative to the base, $d^2 \{\Delta\} dt^2$.

Harmonic motion is assumed and the $\{\Delta\}$ is expressed as:

$$\{\Delta\} = \{\delta\} \sin \omega t \tag{3.7(N)-7}$$

where:

$\{\delta\}$ = column matrix of the spatial displacement and rotation at each mass point relative to the base

ω = natural frequency of harmonic motion in radians per second

The displacement function and its second derivative are substituted into Equation (3.7(N)-6) and yield:

$$[K] \{\delta\} = \omega^2 [M] \{\delta\} \tag{3.7(N)-8}$$

The determinant $\left| [K] - \omega^2 [M] \right|$ is set equal to zero and is then solved for the natural frequencies. The associated mode shapes are then obtained from Equation (3.7(N)-8). This yields n natural frequencies and mode shapes where n equals the number of dynamic degrees of freedom of the system. The mode shapes are all orthogonal to each other and are sometimes referred to as normal mode vibrations. For a single degree of freedom system, the stiffness matrix and mass matrix are single terms and the determinant $\left| [K] - \omega^2 [M] \right|$ when set equal to zero yields simply:

$$k - \omega^2 m = 0 \text{ or:} \tag{3.7(N)-9}$$

$\omega = \sqrt{\frac{k}{m}}$ where ω is the natural angular frequency in radians per second.

The natural frequency in cycles per second is therefore:

$$f = \frac{1}{2\pi} \sqrt{\frac{k}{m}} \tag{3.7(N)-10}$$

To find the mode shapes, the natural frequency corresponding to a particular mode, ω_n, can be substituted in Equation (3.7(N)-8).

2. Modal Equations

The response of a structure or component is always some combination of its normal modes. Good accuracy can usually be obtained by using only the first few modes of vibration. In the normal mode method, the mode shapes are used as principal coordinates to reduce the equations of motion to a set of uncoupled differential equations that describe the motion of each mode n. These equations may be written as (Reference 4, pp. 116 125):

$$\ddot{A}_n + 2\omega_n p_n A + \omega_n^2 A_n = -\Gamma_n \ddot{y}_s \qquad (3.7(N)\text{-}11)$$

where the modal displacement or rotation, A_n, is related to the displacement or rotation of mass point r in mode n, u_{rn}, by the equation:

$$u_{rn} + A_n \varphi_{rn} \qquad (3.7(N)\text{-}12)$$

where:

ω_n = natural frequency of mode n in radians per second.

P_n = critical damping ratio of mode n.

Γ_n = modal participation factor of mode n given by

$$\Gamma_n = \frac{\sum^{n} m_r \varphi'_{rn}}{\sum^{n} m_r \varphi^2_{rn}} \qquad (3.7(N)\text{-}13)$$

where:

φ_{rn} = value of φ_{rn} in the direction of the earthquake.

The essence of the modal analysis lies in the fact that Equation (3.7(N)-11) is analogous to the equation of motion for a single degree of freedom system that will be developed from Equation (3.7(N)-4). Dividing Equation (3.7(N)-4) by m gives:

$$\ddot{u} + \frac{c}{m}\dot{u} + \frac{k}{m}u = -\ddot{y}_s \tag{3.7(N)-14}$$

The critical damping ratio of the single degree of freedom system, p, is defined by the equation:

$$p = \frac{c}{c_c} \tag{3.7(N)-15}$$

where the critical damping coefficient is given by the expression:

$$c_c = 2m\omega \tag{3.7(N)-16}$$

Substituting Equation (3.7(N)-16) into Equation (3.7(N)-15) and solving for c/m gives:

$$\frac{c}{m} = 2\omega p \tag{3.7(N)-17}$$

Substituting this expression and the expression for k/m given by Equation (3.7(N)-9) into Equation (3.7(N)-14) gives:

$$\ddot{u} + 2\omega p\dot{u} + \omega^2 = -\ddot{y}_s \tag{3.7(N)-18}$$

Note the similarity of Equations (3.7(N)-11) and (3.7(N)-18). Thus, each mode may be analyzed as though it were a single degree of freedom system and all modes are independent of each other. By this method a fraction of critical damping, i.e., c/c_c, may be assigned to each mode and it is not necessary to identify or evaluate individual damping coefficients, i.e., c. However, assigning only a single damping ratio to each mode has a drawback. There are three ways used to overcome this limitation when considering a slightly damped structure (e.g., steel) supported by a massive moderately damped structure (e.g., concrete).

The first method is to develop and analyze separate mathematical models for both structures using their respective damping values. The massive moderately damped support structure is analyzed first. The calculated response at the support points for the slightly damped structures is used as a forcing function for the subsequent detailed analysis. The second method is to inspect the mode shapes to determine which modes correspond to the slightly damped structure and then use the damping associated with the structure having predominant motion. The third method is to use the Rayleigh damping method based on computed modal energy distribution.

c. Response Spectrum Analysis

The response spectrum is a plot showing the variation in the maximum response (Reference 4, pages 24 through 51) (displacement, velocity, and acceleration) of a single degree of freedom system versus its natural frequency of vibration when subjected to a time history motion of its base.

The response spectrum concept can be best explained by outlining the steps involved in developing a spectrum curve. Determination of a single point on the curve requires that the response (displacement, velocity, and acceleration) of a single degree of freedom system with a given damping and natural frequency is calculated for a given base motion. The variations in response are established and the maximum absolute value of each is plotted as an ordinate with the natural frequency used as the abscissa. The process is repeated for other assumed values of frequency in sufficient detail to establish the complete curve. Other curves corresponding to different fractions of critical damping are obtained in a similar fashion. Thus, the determination of each point of the curve requires a complete dynamic response analysis, and the determination of a complete spectrum may involve hundreds of such analyses. However, once a response spectrum plot is generated for the particular base motion, it may be used to analyze each structure and component with the base motion. The spectral acceleration, velocity, and displacement are related by the equation:

$$S_{a_n} = \omega_n S_{v_n} = \omega_n{}^2 S_{d_n} \qquad (3.7(N)\text{-}19)$$

There are two types of response spectra that must be considered. If a given building is shown to be rigid and to have a hard foundation, the ground response spectrum or ground time history is used. It is referred to as a ground response spectrum. If the building is flexible and/or has a soft foundation, the ground response spectrum is modified to include these effects. The response spectrum at various support points must be developed. These are called floor response spectra.

d. <u>Integration of Modal Equations</u>

This method can be separated into the following two basic parts:

- Integration procedure for the uncoupled modal Equation (3.7(N)-11) to obtain the modal displacements and accelerations as a function of time.

    - Using these modal displacements and accelerations to obtain the total displacements, accelerations, forces, and stresses.

1. Integration Procedure

    Integration of these uncoupled modal equations is done by step-by-step numerical integration. The step-by-step numerical integration procedure consists of selecting a suitable time interval, Δt, and calculating modal acceleration, $\ddot{A}_n$, modal velocity, $\dot{A}_n$, and modal displacement, A_n, at discrete time stations Δt apart, starting at t = 0 and continuing through the range of interest for a given time history of base acceleration.

2. Total Displacements, Accelerations, Forces and Stresses

    From the modal displacements and accelerations, the total displacements, accelerations, forces, and stresses can be determined as follows:

(a) Displacement of mass point r in mode n as a function of time is given by Equation (3.7(N)-12) as:

$$u_{rn} = A_n \varphi_{rn} \qquad (3.7(N)\text{-}20)$$

with the corresponding acceleration of mass point r in mode n as:

$$u_{rn} = A_n \varphi_{rn} \qquad (3.7(N)\text{-}21)$$

(b) The displacement and acceleration values obtained for the various modes are superimposed algebraically to give the total displacement and acceleration at each time interval.

(c) The total acceleration at each time interval is multiplied by the mass to give an equivalent static force. Stresses are calculated by applying these forces to the model or from the deflections at each time interval.

e. Integration of Coupled Equations of Motion

The dynamic transient analysis is a time history solution of the response of a given structure to known forces and/or displacement forcing functions. The structure may include linear or nonlinear elements, gaps, interfaces, plastic elements, and viscous and Coulomb dampers. Nodal displacements, nodal forces, pressure, and/or temperatures may be considered as forcing functions. Nodal displacements and elemental stresses for the complete structure are calculated as functions of time.

The basic equations for the dynamic analysis are as follows:

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{F(t)\} \qquad (3.7(N)\text{-}22)$$

where the terms are as defined earlier and {F(t)} may include the effects of applied displacements, forces, pressures, temperatures, or nonlinear effects such as plasticity and dynamic elements with gaps. Options of translational accelerations input to a structural system and the inclusion of static deformation and/or pre-load may be considered in the nonlinear dynamic transient analysis. The option of translational input such as uniform base motion to a structural system is considered by introducing an inertia force term of -[M]{z} to the right hand side of the basic Equation (3.7(N)-22), i.e.,

$$[M]\{\ddot{x}\} = [C]\{\dot{x}\} + [K]\{x\} = \{F\} - [M]\{\ddot{z}\} \qquad (3.7(N)\text{-}23)$$

The vector {z} is defined by its components z_i where i refers to each degree of freedom of the system. z_i is equal to a_1, a_2, or a_3 if the i-th degree of freedom is aligned with the direction of the system translational acceleration a_1, a_2, or a_3, respectively. $z_i = 0$ if the i-th degree of freedom is not aligned with any direction of the system translational acceleration. Typical application of this option is a structural system subjected to a seismic excitation of a given ground acceleration record. The displacement {x} obtained from the solution of Equation (3.7(N)-23) is the displacement relative to the ground.

The option of the inclusion of initial static deformation or preload in a nonlinear transient dynamic structural analysis is considered by solving the static problem prior to the dynamic analysis. At each state of integration in transient analysis, the portion of internal forces due to static deformation is always balanced by the portion of the forces which are statically applied. Hence, only the portion of the forces which deviate from the static loads will produce dynamic effects. The output of this analysis is the total result due to static and dynamic applied loads.

One available method for the numerical integration of Equations (3.7(N)-22) and (3.7(N)-23) is the Newmark Beta integration scheme proposed by Chan, Cox, and Benfield (Reference 6). In this integration scheme, Equations (3.7(N)-22) and (3.7(N)-23) are replaced by:

$$\begin{aligned}&\frac{1}{(\Delta t)^2}[M]\{x_{n+2} - 2x_{n+1} + x_n\} + \frac{1}{2(\Delta t)}\{x_{n+2} - x_n\}[C] \\ &\quad + [K]\{\beta x_{n+2} + (1-2\beta)x_{n+1} + \beta x_n\} \\ &\quad = \{\beta F_{n+2} + (1-2\beta)F_{n+1} + \beta x_n\}\end{aligned} \qquad (3.7(N)\text{-}24)$$

where:

n, n+1, n+2	=	past, present, and future (updated) values of the variables
β	=	parameter to be selected on the basis of numerical stability and accuracy
F	=	the total right hand side of the equation of motion (Equation (3.7(N)-22) or (3.7(N)-23))
Δt	=	$t_{n+2} - t_{n+1} = t_{n+1} - t_n$

The value of β is chosen equal to 1/3 in order to provide a margin of numerical stability for nonlinear problems. Since the numerical stability of Equation (3.7(N)-24) is mostly determined by the left hand side terms of that equation, the right hand side terms were replaced by F_{n+2}. Furthermore, since the time increment may vary between two successive time substeps, Equation (3.7(N)-24) may be modified as follows:

$$\begin{aligned}&\frac{2}{(\Delta t + \Delta t_1)}[M]\left\{\frac{x_{n+2} - x_{n+1}}{\Delta t} \quad \frac{x_{n+1} - x_n}{\Delta t_1}\right\} \\ &+ \frac{1}{(\Delta t + \Delta t_1)}[C]\{x_{n+2} - x_n\} + \\ &\tfrac{1}{3}[K]\{x_{n+2} + x_{n+1} + x_n\} = \{F_{n+2}\}\end{aligned} \qquad (3.7(N)\text{-}25)$$

By factoring x_{n+2}, x_{n+1}, and x_n, and rearranging terms, Equation (3.7(N)-25) is obtained as follows:

$$\begin{aligned}&\left\{C_5[M] + C_3[C] + \left(\tfrac{1}{3}\right)K\right\}\{x_{n+2}\} = \{F_n + 2\} \\ &\quad + \left\{C_7[M] - \left(\tfrac{1}{3}\right)[K]\right\}\{x_{n+1}\}\end{aligned} \qquad (3.7(N)\text{-}26)$$

where:

$$C_2 = \frac{2}{\Delta t_1(\Delta t + \Delta t_1)}$$

$$C_3 = \frac{1}{\Delta t + \Delta t_1}$$

$$C_5 = \frac{2}{\Delta t(\Delta t + \Delta t_1)}$$

$$C_7 = C_2 + C_5$$

The above set of simultaneous linear equations is solved to obtain the present values of modal displacements $\{x_t\}$ in terms of the previous (known) values of the nodal displacements. Since [M], [C], and [K] are included in the equation, they can also be time or displacement dependent.

3.7(N).2.2 Natural Frequencies and Response Loads

Refer to Subsection 3.7(B).2.2.

3.7(N).2.3 Procedures Used for Modeling

Procedures used for modeling are discussed in Subsection 3.7(N).2.1a.

3.7(N).2.4 Soil/Structure Interaction

Refer to Subsection 3.7(B).2.4.

3.7(N).2.5 Development of Floor Response Spectra

For Westinghouse floor response spectra development scope, a seismic time history analysis using a coupled building-NSSS model is performed. The seismic input (one input for the OBE analysis and one input for the SSE analysis) is composed of three time history components which are applied simultaneously at the base mat location. The time history input is applied at the base mat location. Using the seismic response from the time history analysis, reactor coolant response spectra are developed as outlined in Subsection 3.7(N).2.1c. The reactor coolant system spectra is used to evaluate feedwater lines, main steam lines, and pressurizer safety and relief valve lines.

3.7(N).2.6 Three Components of Earthquake Motion

The seismic design of the piping and equipment includes the effect of the seismic response of the supports, equipment, structures and components. The system and equipment response is determined using three earthquake components, two horizontal and one vertical. The design ground spectra, are the bases for generating these three input components. Floor response spectra are generated for two perpendicular horizontal directions (i.e., N-S, E-W) and the vertical direction. System and equipment analysis is performed with these input components applied in the N-S, E-W and vertical directions. The damping values used in the analysis are those given in Table 3.7(N)-1.

In computing the system and equipment response by response spectrum modal analysis, the methods of Subsection 3.7(N).2.7 are used to combine all significant modal responses to obtain the combined unidirectional responses.

The combined total response is then calculated using the square-root-of-the-sum-of-the-squares formula applied to the resultant unidirectional responses. For instance, for each item of interest such as displacement, force, stresses, etc., the total response is obtained by applying the above described method. The mathematical expression for this method (with R as the item of interest) is:

$$R_C = \left[\sum_{T=1}^{3} R_T^2 \right]^{1/2} \qquad (3.7(N)\text{-}27)$$

where

$$R_T = \left[\sum_{i=1}^{N} R_{T_i}^2 \right]^{1/2} \qquad (3.7(N)\text{-}28)$$

where

R_C	=	total combined response at a point
R_T	=	value of combined response of direction T
R_{Ti}	=	absolute value of response for direction T, mode i
N	=	total number of modes considered

The subscripts can be reversed without changing the results of the combination.

For the case of closely spaced modes, R_T in Equation (3.7(N)-28) shall be replaced with R_T as given by Equation (3.7(N)-29) in Subsection 3.7(N).2.7.

The system and equipment response can also be determined using time history analyses.

If the time history analysis is performed by applying the two horizontal and vertical time history components independently, the total combined response is computed adding algebraically the unidirectional responses at each time step.

When the three time history components are applied simultaneously, the total response is obtained by direct time integration of the equations of motion.

3.7(N).2.7 Combination of Modal Responses

The total unidirectional seismic response is obtained by combining the individual modal responses utilizing the square-root-of-the-sum-of-the-squares method. For systems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes. The groups of closely spaced modes are chosen such that the difference between the frequencies of the first mode and the last mode in the group does not exceed 10 percent of the lower frequency. Groups are formed starting from the lowest frequency and working towards successively higher frequencies. No one frequency is in more than one group. Combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the square root of the sum of the squares of all modes the product of the responses of the modes in each group of closely spaced modes and a coupling factor $\in$. This can be represented mathematically as:

$$R_T{}^2 = \sum_{i=1}^{N} R_i{}^2 + \sum_{j=1}^{S} \sum_{K=M_j}^{N_{j-1}} \sum_{\ell=K+1}^{N_j} R_K R_\ell \notin_{K\ell} \qquad (3.7(N)\text{-}29)$$

where:

R_T	=	total unidirectional response
R_i	=	absolute value of response of mode i
N	=	total number of modes considered
S	=	number of groups of closely spaced modes
M_j	=	lowest modal number associated with group j of closely spaced modes
N_j	=	highest modal number associated with group j of closely spaced modes
$\in_K$	=	coupling factor with

$$\notin_K = \left\{1, + \left[\frac{{}'\,\omega K'\, {}^{-\omega}\ell}{\left(\beta_K{}^{\omega} K + {}^{\beta}\ell^{\omega}\ell\right)}\right]^2\right\}^{-1} \qquad (3.7(N)\text{-}30)$$

and:

$$\omega'_K = \omega_K\left[1-(\beta'_K)^2\right]^{1/2} \qquad (3.7(N)\text{-}31)$$

$$\beta'_K = \beta_K + \frac{2}{\omega_K t_d} \qquad (3.7(N)\text{-}32)$$

where:

ω_K	=	frequency of closely spaced mode K
β_K	=	fraction of critical damping in closely spaced mode K
t_d	=	duration of the earthquake

An example of this equation applied to a system can be supplied with the following considerations. Assume that the predominant contributing modes have frequencies as given below:

Mode	1	2	3	4	5	6	7	8
Frequency	5.0	8.0	8.3	8.6	11.0	15.5	16.0	20.0

There are two groups of closely spaced modes, namely with modes (2, 3, 4) and (6, 7). Therefore:

S	=	2, number of groups of closely spaced modes
M_1	=	2, lowest modal number associated with group 1
N_1	=	4, highest modal number associated with group 1
M_2	=	6, lowest modal number associated with group 2
N_2	=	7, highest modal number associated with group 2
N	=	8, total number of modes considered

The total response for this system is, as derived from the expansion of Equation (3.7(N)-29):

$$\begin{aligned} R_T^2 = &\left[R_1^2 + R_2^2 + R_3^2 + \cdots + R_8^3\right] + R_2 R_3 \in_{23} + 2R_2 R_4 \in_{24} \\ &+ 2R_3 R_4 \in_{34} + 2R_6 R_7 \in_{67} \end{aligned} \qquad (3.7(N)\text{-}33)$$

3.7(N).2.8 Interaction of Non-Category I Structures with Seismic Category I Structures

Refer to Subsection 3.7(B).2.8.

3.7(N).2.9 Effects of Parameter Variations on Floor Response Spectra

Refer to Subsection 3.7(B).2.9.

3.7(N).2.10 Use of Constant Vertical Static Factors

Constant vertical static factors are not used on the vertical floor response load for the seismic design of safety classed systems and components within Westinghouse's scope of responsibility. All such systems and components are analyzed in the vertical direction.

3.7(N).2.11 Methods Used to Account for Torsional Effects

Refer to Subsection 3.7(B).2.11.

3.7(N).2.12 Comparison of Responses

Refer to Subsection 3.7(B).2.12.

3.7(N).2.13 Methods for Seismic Analysis of Dams

Refer to Subsection 3.7(B).2.13.

3.7(N).2.14 Determination of Seismic Category I Structure Overturning Moments

Refer to Subsection 3.7(B).2.14.

3.7(N).2.15 Analysis Procedure for Damping

In instances under the standard scope of Westinghouse supply and analysis, either the lowest damping value associated with the elements of the system is used for all modes, or an equivalent modal damping value is determined by testing programs such as was done for the reactor coolant loop (Reference 1). Procedures for damping are further discussed in Subsection 3.7(N).2.1.

3.7(N).3 Seismic Subsystem Analysis

This section describes the seismic analysis performed on subsystems within Westinghouse's scope of responsibility.

3.7(N).3.1 Seismic Analysis Methods

Seismic analysis methods for subsystems within Westinghouse scope of responsibility are given in Subsection 3.7(N).2.1.

3.7(N).3.2 Determination of the Number of Earthquake Cycles

For each OBE, the system and components will have a maximum response corresponding to the maximum induced stresses. The effect of these maximum stresses for the total number of OBEs must be evaluated to assure resistance to cyclic loading.

The OBE is conservatively assumed to occur 20 times over the life of the plant. The number of maximum stress cycles for each occurrence depends on the system and component damping values, complexity of the system and component, duration and frequency contents of the input earthquake. A precise determination of the number of maximum stress cycles can only be made using time history analysis for each item, which is not feasible. Instead, a time history study has been conducted to arrive at a realistic number of maximum stress cycles for all Westinghouse systems and components.

To determine the conservative equivalent number of cycles of maximum stress associated with each occurrence, an evaluation was performed considering both equipment and its supporting building structure as single-degree-of-freedom systems. The natural frequencies of the building and the equipment are conservatively chosen to coincide. The damping in the equipment and building are equivalent to the damping values in Table 3.7(N)-1.

The results of this study indicate that the total number of maximum stress cycles in the equipment having peak acceleration above 90 percent of the maximum absolute acceleration did not exceed eight cycles. If the equipment was assumed to be rigid in a flexible building, the number of cycles exceeding 90 percent of the maximum stress was not greater than three cycles.

This study was conservative since it was performed with single-degree-of-freedom models which tend to produce a more uniform and unattenuated response than a complex interacting system. The conclusions indicate that 10 maximum stress cycles for flexible equipment (natural frequencies less than 33 Hz.) and 5 maximum stress cycles for rigid equipment (natural frequencies greater than 33 Hz.) for each of 20 OBE occurrences should be used for fatigue evaluation of Westinghouse systems and components.

3.7(N).3.3 Procedure Used for Modeling

Refer to Subsection 3.7(N).2.1a for modeling procedures for subsystems in Westinghouse scope of responsibility.

3.7(N).3.4 Basis for Selection of Frequencies

The analysis of equipment subjected to seismic loading involves several basic steps, the first of which is the establishment of the intensity of the seismic loading. Considering that the seismic input originates at the point of support, the response of the equipment and its associated supports based upon the mass and stiffness characteristics of the system, will determine the seismic accelerations which the equipment must withstand.

Three ranges of equipment/support behavior which affect the magnitude of the seismic acceleration are possible:

a. If the equipment is rigid relative to the structure, the maximum acceleration of the equipment mass approaches that of the structure at the point of equipment support. The equipment acceleration value in this case corresponds to the low-period region of the floor response spectra.

b. If the equipment is very flexible relative to the structure, the equipment will show very little response.

c. If the periods of the equipment and supporting structure are nearly equal, resonance occurs and must be taken into account.

In all cases, equipment under earthquake loadings is designed to be within code allowable stresses.

Also, as noted in Subsection 3.7(N).3.2, rigid equipment/support systems have natural frequencies greater than 33 Hz.

3.7(N).3.5 Use of Equivalent Static Load Method of Analysis

The static load equivalent or static analysis method involves the multiplication of the total weight of the equipment or component member by the specified seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient is established on the basis of the expected dynamic response characteristics of the component. Components which can be adequately characterized as single-degree-of-freedom systems are considered to have a modal participation factor of one. Seismic acceleration coefficients for multi-degree of freedom systems which may be in the resonance region of the amplified response spectra curves are increased by 50 percent to account conservatively for the increased modal participation and the contributions of the higher modes of vibration.

3.7(N).3.6 Three Components of Earthquake Motion

Methods used to account for three components of earthquake motion for subsystems in Westinghouse's scope of responsibility are given in Subsection 3.7(N).2.6.

3.7(N).3.7 Combination of Modal Responses

Methods used to combine modal responses for subsystems in Westinghouse's scope of responsibility are given in Subsection 3.7(N).2.7.

3.7(N).3.8 Analytical Procedures for Piping

The Class I piping systems are analyzed to the rules of the ASME Code, Section III, NB 3650. When response spectrum methods are used to evaluate piping systems supported at different elevations, the following procedures are used. The effect of differential seismic movement of the piping supports is included in the piping analysis according to the rules of the ASME Boiler and Pressure Vessel Code, Section III, Paragraph NB 3653. According to ASME definitions, these displacements cause secondary stresses in the piping system. The response quantity of interest induced by differential seismic motion of the support is computed statically by considering the building response on a mode-by-mode basis.

In the response spectrum dynamic analysis for evaluation of piping systems supported at different elevations, spectra which envelop the floor response spectra corresponding to the applicable support locations or nozzles are used. Westinghouse does not have in their scope of analysis any piping systems interconnected between buildings.

3.7(N).3.9 Multiply-Supported Equipment Components with Distinct Inputs

When response spectrum methods are used to evaluate reactor coolant system primary components interconnected between floors, the procedure of the following paragraphs is used. The primary components of the Reactor Coolant System are supported at no more than two floor elevations.

A dynamic response spectrum analysis is first made assuming no relative displacement between support points. The response spectra used in this analysis is the most severe floor response spectra.

Secondly, the effect of differential seismic movement of components interconnected between floors is considered statically in the integrated system analysis and in the detailed component analysis. The results of the building analysis are reviewed on a mode-by-mode basis to determine the differential motion in each mode. Per ASME Code rules, the stress caused by differential seismic motion is clearly secondary for piping (NB 3650) and component supports (NF 3231). For components, the differential motion will be evaluated as a free end displacement, since, per NB 3213.19, examples of a free end displacement are motions "that would occur because of relative thermal expansion of piping, equipment, and equipment supports, or because of rotations imposed upon the equipment by sources other than the piping." The effect of the differential motion is to impose a rotation on the component from the building. This motion, then, being a free end displacement and being similar to thermal expansion loads, will cause stresses which will be evaluated with ASME Code methods including the rules of NB 3227.5 used for stresses originating from restrained free end displacements.

The results of these two steps, the dynamic inertia analysis and the static differential motion analysis, are combined absolutely with due consideration for the ASME classification of the stresses.

3.7(N).3.10 Use of Constant Vertical Static Factors

Constant vertical load factors are not used on the vertical floor response load for the seismic design of safety-related components and equipment within Westinghouse's scope of responsibility.

3.7(N).3.11 Torsional Effects of Eccentric Masses

The effect of eccentric masses, such as valves and valve operators, is considered in the seismic piping analyses. These eccentric masses are modeled in the system analysis and the torsional effects caused by them are evaluated and included in the total system response. The total response must meet the limits of the criteria applicable to the safety class of the piping.

3.7(N).3.12 Buried Seismic Category I Piping Systems and Tunnels

Refer to Subsection 3.7(B).3.12.

3.7(N).3.13 Interaction of Other Piping with Seismic Category I Piping

Refer to Subsection 3.7(B).3.13.

3.7(N).3.14 Seismic Analysis for Reactor Internals

Fuel assembly component stresses induced by horizontal seismic disturbances are analyzed through the use of finite element computer modeling. The time history floor response based on a standard seismic time history normalized to SSE levels is used as the seismic input. The reactor internals and the fuel assemblies are modeled as spring and lumped mass systems or beam elements. The component seismic response of the fuel assemblies is analyzed to determine design adequacy. A detailed discussion of the analyses performed for typical fuel assemblies is contained in Reference 7.

Fuel assembly lateral structural damping obtained experimentally is presented in Reference 7 (Figure B-4). The distribution of fuel assembly amplitudes decreases as one approaches the center of the core. Fuel assembly displacement time history for the SSE seismic input is illustrated in Reference 7 (Figure 2-3).

The CRDMs are seismically analyzed to confirm that system stresses under the combined loading conditions, as described in Subsection 3.9(N).1, do not exceed allowable levels, as defined by the ASME Code, Section III for "Upset" and "Faulted" conditions. The CRDM is mathematically modeled as a system of lumped and distributed masses. The model is analyzed under appropriate seismic excitation, and the resultant seismic bending moments along the length of the CRDM are calculated. The corresponding stresses are then combined with the stresses from the other loadings required, and the combination is shown to meet the ASME Code, Section III, requirements.

3.7(N).3.15 Analysis Procedure for Damping

Analysis procedures for damping for subsystems in Westinghouse's scope of responsibility are given in Subsection 3.7(N).2.15.

3.7(N).4 Seismic Instrumentation

Refer to Subsection 3.7(B).4.

3.7(N).5 References

1. "Damping Values of Nuclear Plant Components," WCAP-7921-AR, May 1974.

2. Geninski, T. L. and Chiang, D., "Safety Analysis of the 17x17 Fuel Assembly for combined Seismic and Loss-of-Coolant Accident," WCAP-8236 (Proprietary), December 1973 and WCAP-8286 (Nonproprietary), December 1974.

3. Lin, C. W., "How to Lump the Masses - A Guide to the Piping Seismic Analysis," ASME paper 74-NE-7 presented at the Pressure Vessels and Piping Conference, Miami, Florida, June 1974.

4. Biggs, J. M., Introduction to Structural Dynamics, McGraw-Hill, New York, 1964.

5. Thomas, T. H., et al., "Nuclear Reactors and Earthquakes," TID-7024, U.S. Atomic Energy Commission, Washington, D. C., August 1963.

6. Chan, S. P., Cox, H. L. and Benfield, W. A., "Transient Analysis of Forced Vibration of Complex Structural-Mechanical Systems," J. Royal Aeronautical Society," July 1962.

7. "Safety Analysis of the 8-Grid 17x17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident," WCAP-8236, Addendum 1 (Proprietary), March 1974 and WCAP-8288, Addendum 1 (Non-proprietary), April 1974.

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 Concrete Containment

The containment structure houses the major portion of a PWR Nuclear Steam Supply System (NSSS). During the operating life of the plant, it will also provide the following functions:

a. Limiting the leakage rate to the maximum allowable Type "A" test leakage rate, 0.15 percent by weight of the containment contained air mass per day at calculated peak pressure and associated temperature, resulting from any loss-of-coolant accident (LOCA) and other postulated accidents.

b. Providing continuing radiation shielding during normal plant operation in accordance with 10 CFR 20 and during accident conditions in accordance with 10 CFR 100.

c. Protecting the reactor vessel and all other safety-related systems, equipment and components located inside the containment against all postulated external environmental conditions and resulting loads.

3.8.1.1 Description of Containment

The containment, Figure 1.2-2, Figure 1.2-3, Figure 1.2-4, Figure 1.2-5 and Figure 1.2-6, is a seismic Category I reinforced concrete dry structure, which is designed to function at atmospheric conditions. It consists of an upright cylinder topped with a hemispherical dome, supported on a reinforced concrete foundation mat which is keyed into the bedrock by the depression for the reactor pit and by continuous bearing around the periphery of the foundation mat. The inside diameter of the cylinder is 140 feet and the inside height from the top of the base mat to the apex of the dome is approximately 219 feet; the net free volume is approximately 2,704,000 cubic feet.

A welded steel liner plate, anchored to the inside face of the containment, serves as a leaktight membrane. Although not a code requirement, welds that are embedded in concrete and not readily accessible are covered by a leak chase system which permits leak testing of those welds throughout the life of the plant. Exemptions to these inaccessible welds are the welds joining mechanical penetrations X-60 and X-61 to the steel liner plate. (The venting pipes which join the leak chase channels for these penetrations to the atmosphere were not provided; however, these welds underwent proper testing before they became inaccessible.) The liner on top of the foundation mat is protected by a four feet thick concrete fill mat which supports the containment internals and forms the floor of the containment.

The containment is designed to assure that the base mat, cylinder, and dome behave integrally to resist all loads.

Located outside the Containment Building and having a similar geometry is the Containment Enclosure Building. This structure provides leak protection for the containment and protects it from certain loads, as discussed in Subsection 3.8.1.3. The Containment Enclosure Building is described in Subsection 3.8.4.

a. Base Mat

The reinforced concrete base mat is 153 feet in diameter and 10 feet thick. It is designed to carry the loads from the shell of the containment and from the internal structures.

An orthogonal grid rebar arrangement is provided for the bottom face of the base mat to simplify fabrication and construction. A radial and hoop pattern is used at the top face to minimize interference with cylinder dowels. Vertical and inclined shear reinforcement are provided to resist the transverse shear forces caused by design accident pressure and seismic loads. Details of the base mat reinforcing steel are shown on Figure 3.8-1. The mat liner plate is 1/4" thick with joints welded to leveling angles which serve as welding backup strips.

Internal structures are supported on and anchored to the fill mat, as indicated above. The mat is not anchored to the base mat. Stability of the containment with internals is provided by the keying action of the base mat and reactor pit in the rock and by bearing against the foundation for the Enclosure Building, which in turn transfers all horizontal shears directly into the bedrock through fill concrete.

b. Cylinder

The cylinder has an inside diameter of 140 feet and is nominally 4'-6" thick. Also, it is thickened to provide room for additional reinforcing steel around the openings for the equipment hatch and the personnel air lock.

The reinforcing bars in the cylinder are arranged and oriented to resist hoop, meridional and shear forces, including hoop, meridional and radial shear forces produced by bending moments. Orthogonal layers of bars in the meridional and hoop directions are provided on each face to resist the membrane forces primarily from pressure and seismic loads.

An orthogonal set of bars inclined at 45 degrees to the horizontal is provided on the outside face to resist in-plane seismic shear forces and membrane tension from other loads. Near the base of the containment, additional meridional bars and radial inclined stirrups are provided to resist discontinuity moments and radial shears, respectively, caused by the restraint on the cylinder at the junction of the cylinder and the base mat. Stirrups are also provided at the springline to resist radial shear.

Where there are large openings for access hatchways and penetrations, the main reinforcing bars are continued without interruption around the openings. No main reinforcing bars are terminated at any opening. Furthermore, additional bars are provided to resist the local effects of these openings and, around large openings such as the equipment hatch (28'-0" inside diameter) and personnel air lock (7'-1¼" inside diameter), the concrete is thickened locally to resist the additional local forces and to accommodate the additional reinforcing.

Basic details of reinforcing steel in the cylinder are shown on Figure 3.8-2, which also includes reinforcing steel in the dome and junction of the wall and base mat. Details of the reinforcing steel at the equipment hatch and personnel air lock are shown on Figure 3.8-3 and Figure 3.8-4, respectively. These figures also show the transition of reinforcing steel between the openings and the membrane regions.

The liner plate in the cylinder is 3/8" thick in all areas except penetrations and the junction of the base mat and cylinder where it is 3/4" thick. The liner is provided with an anchorage system to assure that it can withstand accident loadings while maintaining leak tightness. In addition, the anchorage system assures that the liner, which is used as a form during construction, can resist the hydrostatic concrete loads while maintaining liner tolerances within allowable values. The anchorage system consists of vertical tees spaced every 20 inches around the circumference of the cylindrical wall. The webs of the tees are welded to the liner plate with two ¼" continuous fillet welds. Bent studs are attached to the flange of vertical tees as required to accommodate placement of rebar. Liner details are shown on Figure 3.8-5.

Containment penetrations, other than the equipment hatch and personnel air lock, are located in the lower portion of the cylindrical structure. In general, a penetration consists of a sleeve anchored in the concrete cylinder wall and welded to the locally thickened containment liner. The weld between the liner and the sleeve is covered by a leak chase system which can be pressurized to demonstrate the integrity of the penetration-to-liner weld joint. The piping, electrical cable and instrumentation cable pass through the embedded sleeves and the ends of the resulting annuli are closed off either by welded end plates or a flued head welded to the sleeve outside the containment. If the pipe carries hot fluid, the space between the pipe and the sleeve is insulated to maintain the concrete temperature adjoining the embedded sleeve at or below 200° F during normal plant operation. The fuel transfer tube passes through an embedded sleeve which has its ends closed off by an expansion bellows and an end plate. In the case of ventilation ducts, the sleeve forms the wall of the duct.

Sleeves for all penetrations, including the equipment hatch and personnel air lock, are embedded in the concrete wall by an engineered anchorage system that is welded to the penetration sleeve. Reinforcing steel, hoop, meridional and diagonal, is splayed around penetrations permitting all bars to be continuous. See Figure 3.8-6 for details of reinforcing at penetrations.

All brackets and attachments are welded to attachment plates which are welded to the liner plate and anchored into the concrete by studs welded on the opposite side of the liner, thereby transmitting the forces directly to the concrete. See Figure 3.8-5. In the reactor cavity pit, brackets are used, as a construction aid, on the concrete side of the liner to temporarily support the reinforcing steel. This is a permanent attachment which only functions until the concrete is placed. Each of these brackets is welded to an attachment plate; the attachment plate is welded to the liner with a continuous fillet weld and is not backed by a stud.

c. Dome

The dome is a reinforced concrete shell 3'-6" thick and 69'-11" in radius. Due to the change in concrete thickness, the discontinuity of concrete at the springline is on the outer surface.

Reinforcing steel in the dome consists of hoop, meridional, and diagonal bars, as in the cylinder. The meridional and diagonal bars are continuous with those in the cylinder.

One-half of the meridional bars, in an alternating fashion, are terminated at 60° above the springline with the remaining bars evenly spread and continued across the upper 30° of the dome.

The hoop bars in the dome are terminated where no longer needed at 75° above the springline.

One-half of the diagonal bars are terminated where no longer needed, at approximately 30° above the springline; the remaining bars are terminated at approximately 45° above the springline.

See Figure 3.8-7 for details of the reinforcing steel.

The dome liner is 1/2" thick and flush with the outside face of the cylindrical liner. The anchorage system consists of tees on a 5'-0" grid pattern. A bent stud is located in the center of each of the resulting 5'-0" x 5'-0" panels to provide some additional anchorage. See Figure 3.8-5.

d. Steel Components

Steel components that resist pressure and are not backed by structural concrete include the following:

1. Equipment hatch
2. Personnel air lock
3. High energy piping penetrations
4. Moderate energy piping penetrations
5. Electrical penetrations
6. Fuel transfer tube assembly
7. Instrumentation penetrations
8. Ventilation penetrations

These are discussed in Subsection 3.8.2.

3.8.1.2 Applicable Codes, Standards and Specifications

The design, materials, fabrication, construction, testing and inspection of the containment structure conform to the applicable sections of the following codes and specifications which are used to establish design bases and methods, analytical techniques, material properties and quality control provisions.

Dates and revision given for the listed codes are of the earliest version that was used. Subsequent issues were incorporated into the design where practicable or where the new issue directly affected the safety of the structure.

Any exceptions to the indicated issue of the ASME B&PV Code, Section III, Division 2, are indicated in the text, either in Subsection 3.8.1.4 or in Subsection 3.8.1.5, as appropriate.

Code or Specification	Title
ACI 211.1-74	Recommended Practice for Selecting Proportions For Normal Weight Concrete
ACI 214-65	Recommended Practice for Evaluation of Compression Test Results of Field Concrete
ACI 301-72	Structural Concrete for Building
ACI 304-73	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
ACI Committee Report 74-33	Recommended Practice for Hot Weather Concreting
ACI Report 306R-78	Recommended Practice for Cold Weather Concreting
ACI 308-71	Recommended Practice for Curing Concrete
ACI 309-72	Recommended Practice for Consolidating Concrete
ACI 311-64	Recommended Practice for Concrete Inspections
ACI 315-65	Manual of Standard Practice for Detailing Reinforced Concrete Structures
ACI 318-71	Building Code Requirements for Reinforced Concrete (with Commentary)
ACI 347-68	Recommended Practice for Concrete Formwork
ACI 614-59	Recommended Practice for Measuring, Mixing and Placing Concrete
ACI Committee Report 68-33	Placing Concrete by Pumping Methods

Code or Specification	Title
ACI Committee Report 72-33	Placing Concrete with Belt Conveyors
ACI SP2	Manual of Concrete Inspection, 1975 Edition
CRSI	Reinforced Concrete - Manual of Standard Practice, 22nd Edition, first printing, 1976
CRSI	Recommended Practice for Placing Reinforcing Bars, 1968 Edition
ASME	ASME Boiler and Pressure Vessel Code, Section III, Division 2, Code for Concrete Reactor Vessels and Containments: Concrete Containment Component - 1975 Edition Containment Liner - 1975 Edition (1976 Winter Addendum used for liner allowable stresses) Reinforced Concrete - 1975 Edition through Winter 1976 Addenda (also 1977 Winter Addendum, Article CC-3422 and 1979 Summer Addendum, Article CC-3422) SIT - 1980 Edition, as referenced in UFSAR Subsection 3.8.1.7a, except as noted in applicable subsections of UFSAR Section 3.8.[1] [2]
ASME	ASME Boiler and Pressure Vessel Code, Section II, Material Specification, Part C, Welding Rods, Electrodes and Filler Metals (up to and including Winter 1975 Addenda)
ASME	ASME Boiler and Pressure Vessel Code, Section IX, Welding and Brazing Qualifications (up to and including Winter 1974 Addenda)

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[1] From here on referred to as Division 2.

[2] Article CA-8000 of Division 2 applies, except that in lieu of code symbol application a New Hampshire State special waiver, dated March 18, 1976, regarding marking, stamping and recording has been granted.

Code or Specification	Title
ASME	ASME Boiler and Pressure Vessel Code, Section V, Nondestructive Examination (up to and including 1974 Summer Addendum)
ASME	ASME Boiler and Pressure Vessel Code, Section III, Division 2 Code Cases: Code Case N-218, Testing Lots of Carbon Steel Solid, Base Welding Electrode on Wire, August 14, 1981. Code Case N-219, Rules for Design of Peripheral Shear Reinforcing, January 8, 1979. Code Case N-287, Rules for Design of Radial Shear Reinforcing, July 14, 1980. Code Case N-232, Alternate Rules for Development Length of Reinforcing Steel Not Required to Carry Load, January 21, 1982.
ASME	ASME Boiler and Pressure Vessel Code, Section II, Material Specification, Part A, Ferrous Materials (up to and including Winter 1974 Addenda)
SA-36	Specification for Structural Steel
SA-240	Specification for Stainless and Heat Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet and Strip for Fusion-Welded Unfired Pressure Vessels
SA-300	Specification for Notch Toughness Requirements for Normalized Steel Plates for Pressure Vessels
SA-333	Specification for Seamless and Welded Steel Pipe for Low Temperature Service
SA-442	Specification for Carbon Steel Plates with Improved Transition Properties for Pressure Vessels
SA-516	Specifications for Carbon Steel Plates for Pressure Vessels for Moderate and Lower Temperature Service

Code or Specification	Title
ASTM A36-70	Specification for General Requirements for Delivery of Rolled Steel Plates, Shapes, Sheet Piling and Bars for Structural Use
ASTM A184-65	Specification for Fabricated Steel Bar or Rod Mats for Concrete Reinforcement
ASTM A185-70	Specification for Welded Steel Wire Fabric for Concrete Reinforcement
ASTM A325-71	Specification for High Strength Bolts for Structural Steel Joints, Including Suitable Nuts and Plain Hardened Washers
ASTM A370-75a	Methods and Definitions for Mechanical Testing of Steel Products
ASTM A449-68	Specification for Quenched and Tempered Steel Bolts for Structural Steel Joints
ASTM A490-71	Specification for Quenched and Tempered Alloy Steel Bolts for Structural Steel Joints
ASTM A501-71	Specification for Hot Formed Welded and Seamless Carbon Steel Structural Tubing
ASTM A514-70	Specification for High-Yield Strength, Quenched and Tempered Alloy Steel Plate, Suitable for Welding
ASTM A519-75	Standard Specification for Seamless and Alloy Mechanical Tubing
ASTM A615-75	Specification for Deformed Billet-Steel Bars for Concrete Reinforcement
ASTM C29-71	Standard Methods of Test for Unit Weight of Aggregate
ASTM C31-69	Standard Method of Making and Curing Concrete Compressive and Flexural Strength Test Specimen in the Field
ASTM C33-71a	Specification for Concrete Aggregates
ASTM C39-71	Standard Method of Test for Compressive Strength of Molded Concrete Cylinders

Code or Specification	Title
ASTM C40-73	Standard Method of Test for Organic Impurities in Sands of Concrete
ASTM C42-68	Standard Method of Obtaining and Testing Drilled Cores and Sawed Beams of Concrete
ASTM C70-73	Standard Method of Test for Surface Moisture in Fine Aggregate
ASTM C87-69	Standard Method of Test for Effect of Organic Impurities in Fine Aggregate on Strength of Mortar
ASTM C88-73	Standard Method of Test for Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate
ASTM C109-73	Standard Method of Test for Compressive Strength of Hydraulic Cement Mortars (Using 2-inch Cube Specimens)
ASTM C114-69	Standard Methods of Chemical Analysis of Hydraulic Cement
ASTM C117-69	Standard Method of Test for Materials Finer than No. 200 (75-uM) Sieve in Mineral Aggregates by Washing
ASTM C123-69	Standard Method of Test for Light Weight Pieces in Aggregate
ASTM C125-74	Standard Definitions of Terms Relating to Concrete and Concrete Aggregates
ASTM C127-73	Standard Method of Test for Specific Gravity and Absorption of Coarse Aggregate
ASTM C128-73	Standard Method of Test for Specific Gravity and Absorption of Fine Aggregate
ASTM C131-69	Standard Method of Test for Resistance to Abrasion of Small Size Coarse Aggregate by the Use of the Los Angeles Machine
ASTM C136-71	Standard Method of Test for Sieve of Screen Analysis of Fine and Coarse Aggregates
ASTM C138-75	Standard Method of Test for Unit Weight, Yield, and Air Content (Gravimetric) of Concrete

Code or Specification	Title
ASTM C142-71	Standard Method of Test for Clay Lumps and Friable Particles in Aggregates
ASTM C143-71	Standard Method of Test for Slump of Portland Cement Concrete
ASTM C150-71	Specification for Portland Cement
ASTM C151-74a	Standard Method of Test for Autoclave Expansion of Portland Cement
ASTM C172-71	Standard Method of Sampling Fresh Concrete
ASTM C173-75	Standard Method of Test for Air Content of Freshly Mixed Concrete by the Volumetric Method
ASTM C192-69	Standard Method of Making and Curing Concrete Test Specimens in the Laboratory
ASTM C231-71T	Tentative Method of Test for Air Content of Freshly Mixed Concrete by the Pressure Method
ASTM C233-73	Standard Method of Testing Air-Entraining Admixtures for Concrete
ASTM C235-68	Standard Method of Test for Scratch Hardness of Coarse Aggregate Particles
ASTM C260-69	Specification for Air-Entraining Admixtures for Concrete
ASTM C295-65	Recommended Practice for Petrographic Examination of Aggregates for Concrete
ASTM C309-74	Standard Specification for Liquid Membrane-Forming Compounds for Curing Concrete
ASTM C494-71	Specification for Chemical Admixtures for Concrete
ASTM C496-71	Standard Method of Test for Splitting Tensile Strength of Molded Concrete Cylinders

Code or Specification	Title
ASTM C566-67	Standard Method of Test for Total Moisture Content of Aggregate by Drying
ASTM C666-75	Standard Method of Test for Resistance of Concrete to Rapid Freezing and Thawing
ASTM D75-71	Standard Method of Sampling Aggregates
ASTM D297-74	Methods of Chemical Analysis of Rubber Products
ASTM D412-68	Method of Tension Testing of Vulcanized Rubber
ASTM D573-67	Test for Accelerated Aging of Vulcanized Rubber by the Oven Method
ASTM D624-73	Tests for Tear Resistance of Vulcanized Rubber
ASTM D746-73	Test for Brittleness Temperature of Plastics and Elastomers by Impact
ASTM D816-55	Standard Method of Testing Rubber Cements
ASTM D1149-64	Test for Accelerated Ozone Cracking of Vulcanized Rubber
ASTM D2240-75	Test for Indentation Hardness of Rubber and Plastics by Means of a Durometer
ASTM E96-66	Standard Methods of Test for Water Vapor Transmission of Materials in Sheet Form
ASTM E329-72	Recommended Practice for Inspection and Testing Agencies for Concrete as Used in Construction (Articles 7, 8 and 9 of ASTM E329 shall not apply).
AISC	Code of Standard Practice for Steel Buildings and Bridges (July 1, 1970)
AISC	Specification for the Design, Fabrication and Erection of Structural Steel for Buildings 1969 Edition (including supplements 1, 2 and 3)

Code or Specification	Title
AISC	Specification for Design of Light Gage Cold-Formed Steel Structural Members, 1977
Research Council for Riveted and Bolted Structural Joints of the Engineering Foundation	Specification for Structural Joints Using ASTM A325 or A490 Bolts (September 1, 1966)
ANS 7.60-1972	Standard for Leakage Rate Testing of Containment Structures for Nuclear Reactors
ANSI N177	Plant Design Against Missiles (Draft Standard, April, 1974)
ANSI A58.1-1972	Building Code Requirements for Minimum Design Loads in Buildings and Other Structures
ANSI N45.2-1974	Quality Assurance Program Requirements for Nuclear Power Plants
ANSI N45.3-1973	Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations
ANSI N45.4-1972	Leakage Rate Testing of Containment Structures for Nuclear Reactors
ANSI N101.2-1972	Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities
ANSI N101.4-1972	Quality Assurance for Protective Coating Applied to Nuclear Facilities
ANSI N512-1974	Protective Coatings (Paints) for the Nuclear Industry
AWS D1.0-69	Standard Code for Arc and Gas Welding in Building Construction
AWS B3.0-41	Standard Qualification for Procedures
AWS D12.1-75	Recommended Practice for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction

Code or Specification	Title
Uniform Building Code	International Conference of Building Officials, Uniform Building Code, 1973 Edition
U.S. Dept. of Labor OSHA	Occupational Safety and Health Administration Standards, October, 1975 Edition
U.S. Dept. of Labor QQ-C-40	Caulking: Lead Wool and Lead Pig, 1963
NRC TID 7024	Nuclear Reactors and Earthquakes (Prepared by Lockheed Aircraft Corp. for NRC, August 1963)
PS-1-74	U.S. Dept. of Commerce, National Bureau of Standards, Construction and Industrial Plywood
Steel Structures Painting Manual Vol. 1	Steel Structures Painting Council, Good Painting Practice, 1966
Steel Structures Painting Manual Vol. 2	SSPC, Systems and Specifications, 1973
Federal Specifications and Standards Handbook 44	National Bureau of Standards Specifications, Tolerances and Other Technical Requirements for Commercial Weighing and Measuring Devices, 1971

U.S. Army Corps of Engineer Specifications	Title
CRD-C38-73	Method of Temperature Rise in Concrete
CRD-C39-55	Coefficient of Thermal Expansion
CRD-C44-63	Coefficient of Thermal Conductivity
CRD-C119-53	Method of Test for Flat and Elongated Particles in Coarse Aggregate
CRD-C-588-78A	Specification for Expansive Grouts
CRD-C-589-70	Methods of Sampling and Testing Expansion Grouts

NRC Regulatory Guides	Title
1.10, Rev. 1, 1/73	Mechanical (Cadweld) Splices in Reinforcing Bars of Concrete Containments
1.15, Rev. 1, 12/72	Testing of Reinforcing Bars for Concrete Structures
1.18, Rev. 1, 12/72	Structural Acceptance Test for Concrete Primary Reactor Containments
1.19, Rev. 0, 8/72	Nondestructive Examination of Primary Containment Liners
1.54, Rev. 0, 6/73	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
1.55, Rev. 0, 6/73	Concrete Placement in Category I Structures
1.57, Rev. 0, 6/73	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components
1.60, Rev. 1, 12/73	Design Response Spectra for Seismic Design of Nuclear Power Plants
1.61, Rev. 0, 10/73	Damping Values for Seismic Design of Nuclear Power Plants
1.76, Rev. 0, 4/74	Design Basis Tornado for Nuclear Power Plants
1.84, Rev. 15. 5/79	Code Case Acceptability ASME Section III Design and Fabrication
1.85, Rev. 15, 5/79	Code Case Acceptability ASME Section III Materials
1.92, Rev. 1, 2/76	Combining Modal Responses and Spatial Components in Seismic Response Analysis
1.94, Rev. 1, 4/76	Quality Assurance Requirements for Installation and Testing of Structural Concrete and Structural Steel During Construction Phase of Nuclear Power Plants
1.136, Rev. 2, 6/81	Materials, Construction, and Testing of Concrete Containments

Code of Federal Regulations	Title
10 CFR 20	Standards for Protection Against Radiation
10 CFR 50 App. A	General Design Criteria for Nuclear Power Plants
10 CFR 50 App. B	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants
10 CFR 50 App. J	Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors
10 CFR 100 App. A	Reactor Site Criteria

The below listed UE&C design and construction specifications applicable to the containment structure were prepared in accordance with applicable codes, quality control requirements and NRC Regulatory Guides:

UE&C Specifications	Title
9763.006-1-1	General Concrete Construction, Steel Erection, and Circulating Water Pipe Installation
9763.006-5-1	Civil Testing Facility and Services
9763.006-5-4	Containment Initial Integrated Leakage Rate Test (ILRT)
9763.006-5-5	Containment Structural Integrity Test (SIT)
9763.006-10-1	Dewatering
9763.006-11-1	Foundation Waterproofing
9763.006-13-2	Containment Concrete Work
9763.006-14-1	Furnishing, Detailing, Fabricating, and Delivering Reinforcing Bars
9763.006-14-2	Installation of Reinforcing Bars in Containment Structure
9763.006-15-1	Containment Liner
9763.006-15-2	Containment Equipment Hatch and Personnel Locks

UE&C Specifications	Title
9763.006-18-2	Installation of Miscellaneous Embedded Steel and Weldments
9763-006-18-4	Furnishing and Installing Embedded Steel and Miscellaneous Steel
9763.006-41-4	Furnishing of Protective Coating (Paint) System Materials and Related Services
9763.006-41-7	Field Painting of Containment Structure Interior
9763.006-69-1	Concrete Batch Plant
9763.006-69-3	Concrete Mixes
9763.006-69-7	Standard Concrete Mixes
9763.006-80-1	Containment Design
9763.006-80-2	Construction of Containments
9763-MPS-1	Material and Processing Requirements for Nuclear Power Plant Components
9763-MPS-3	Material and Processing Requirements for Bending of Welded Studs, Reinforcing Bars and Anchor Bolts
9763-QAS-1	Quality Assurance Administrative and System Requirements (Nuclear)
9763-RM-1	Instructions for Site Records Management System
9763-SD-15-2	Seismic Requirements
9763-WS-1	Requirements for Welding and Nondestructive Examination for Nuclear Pressure Components and Nuclear Power Piping
9763-WS-1-NE	Requirements for Welding and Nondestructive Examination for Nuclear Pressure Class MC Components
9763-WS-4A	Requirements for Welding and Nondestructive Examination for Nuclear Containment Structure Liner

UE&C Specifications	Title
9763-WS-4B	Requirements for Stud Welding and Nondestructive Examination for Nuclear Containment Structure Liner
9763-WS-4C	Requirements for Mechanical Splicing and Nondestructive Examination of the Reinforcing Bars Spliced by the Cadweld Method

3.8.1.3 Loads and Loading Combinations

The containment is designed to withstand all credible conditions of loading, including preoperational test loads, normal loads, severe environmental loads, extreme environmental loads, and abnormal loads. These loads are determined in accordance with Article CC-3000 of Division 2 and are considered in the applicable service and factored load combinations to assure that the response of the structure will remain within the design limits prescribed in Subsection 3.8.1.5. Site-related loads are also considered.

a. Design Loads

The definitions of the loads used in the design of the containment include the following:

1. Preoperational Test Loads

Test loads are those which are applied during the initial and any subsequent structural integrity or leak rate testing of the containment. The definitions for dead loads and live loads are those given in the Normal Load section. The following loads are also considered:

(a) Test Pressure (P_t)

The containment is pressurized to 115 percent of the design pressure to test its structural integrity; i.e., test pressure is 60 psig with design pressure being 52 psig.

(b) Test Temperature (T_t)

The thermal loads occurring during the structural integrity test conditions of the containment are induced by the temperature gradient between the inside containment air temperature and the outside air temperature. The maximum and minimum temperatures considered inside the containment during the test are 100° F and 50° F, respectively. During this test, temperature outside the containment (within the Enclosure Building) will be monitored and controlled to maintain a minimum temperature of 30°F; also, the average differential temperature between the inside and outside of the containment does not exceed 65° F, except the equipment hatch and personnel air locks.

2. Normal Startup, Operational, and Shutdown Loads

Normal loads are those loads encountered during normal plant startup, operation, and shutdown. They include the following:

(a) Dead Loads (D)

Dead loads are all permanent gravity loads including, but not limited to, the weight of the base mat, cylindrical wall, dome, internal concrete structures including fill mat, steel liner plate and structural framing, equipment, piping, cable and cable trays and miscellaneous building loads within the containment. Gravity loads from the internal structures are transmitted to the base mat through the fill mat. Included as a dead load is the buoyant force of the groundwater at El. +20'-0" (i.e., 50'-0" above the top of the base mat) as described in Subsection 2.4.13.5. Buoyancy from the hydrostatic loading is considered in the base mat design and containment stability analysis.

(b) Live Loads (L)

Live loads are those loads which vary in intensity and/or occurrence. During normal operation there are no significant live loads on the external containment surface since the Containment Enclosure Building resists snow loads and lateral loads from soil pressure and normal wind. Live loads from the internal structures are transmitted to the base mat through the fill mat and have no significant effect on the containment shell or base mat design.

(c) Operational Thermal Loads (T_o)

The thermal loads experienced during normal operating or shutdown conditions are generated by temperature gradients through the containment shell and by liner expansion. The gradient through the shell during these normal operating conditions varies between 120°F on the inside and -10°F on the outside. In the base mat the gradient varies between approximately 97°F on the inside to a constant 40°F at the interface with the bedrock; the top of the fill mat is at 120°F. These gradients are shown on Figure 3.8-8. Other temperature gradients, such as 50°F on the inside varying to 90°F on the outside may occur, but the loads produced by them do not control the design of the concrete containment. Note also that the effect of the Containment Enclosure Building in reducing the lower bound outside temperatures was not considered in the containment design; that is, -10°F was used in the evaluation of thermal loads.

(d) Operational Pipe Reactions (R_o)

Piping reactions transmitted to the containment during normal operation or shutdown conditions are based on the most critical transient or steady-state condition. The magnitudes of these loads are determined by the piping design and are included in the Design Report. (The Seabrook Station Containment Design Report is prepared in accordance with Subarticle CA-3240 of Division 2 and is retained by the Owner in accordance with the requirements of Subarticles CA-3100 and CA-4832.)

(e) Pressure Variation (P_V)

Differential pressure loads result from pressure variation either inside the containment or in the annulus between the containment and the Containment Enclosure Building. This pressure variation is produced either by atmospheric fluctuations or by HVAC equipment. The containment structure was designed to withstand a maximum external pressure of 3.5 psi (differential).

The stability of the shell was investigated under the effects of this external pressure load.

(f) Alkali Silica Reaction Loads (S_a)

These are structural effects caused by ASR expansion of concrete. ASR loads are passive and therefore occur during normal operation, shutdown conditions, and concurrently with all extreme environmental loads. For Containment, only the effects of ASR expansion occurring in reinforced concrete structural members are considered. (Expansion of concrete backfill is not considered as the concrete backfill does not interact directly with Containment.)

Calculation of the ASR demands are described below; detailed guidance on calculation of the loads is provided in "Methodology for the Analysis of Seismic Category I Structures' with Concrete Affected by Alkali-Silica Reaction," (FP# 101196).

Demands associated with internal ASR expansion are applied to structural components as strain loads in the concrete model based on in-plane expansion measurements. The internal ASR expansion is applied uniformly through the cross-sectional thickness of the structural components (e.g., walls, slabs, foundations, etc.) unless otherwise justified. Application of ASR expansion to the concrete elements that are restrained by reinforcement elements results in compression of the concrete and tension in the reinforcement.

The in-plane ASR expansions can be adjusted when some or all of the cracks at an ASR monitoring grid are shown to be caused by a mechanism other than internal ASR in the reinforced concrete member (e.g., shrinkage, thermal, pressurization tests of Containment, etc.). The adjusted in-plane expansion values are computed by excluding the widths of cracks determined not to be caused by ASR.

3. Severe Environmental Loads

Severe environmental loads are those loads that would result from external conditions which could infrequently be encountered during the plant life. The following loads are included in this category:

(a) Wind Load (W)

There is no wind load considered in the containment design due to the presence of the containment enclosure.

(b) Operating Basis Earthquake Loads (E_o)

These are the loads generated by the Operating Basis Earthquake, which is the earthquake that could reasonably be expected to occur at the plant site during the operating life of the plant. Only the actual dead load and weights of fixed equipment are considered in evaluating the seismic response forces. The horizontal and vertical design response spectra for the OBE are derived by applying a factor of 0.5 to the response spectra given for Safe Shutdown Earthquake (SSE), which is described below. The effects of two (2) orthogonal horizontal components and one (1) vertical component of earthquake are considered and combined by the square-root-of-the-sum-of-the-squares method. Due to the presence of the Containment Enclosure Building, there are no dynamic effects of soil on the containment.

4. Extreme Environmental Loads

Extreme environmental loads are those loads which result from postulated events which are credible, but highly improbable. The following loads are included in this category:

(a) Safe Shutdown Earthquake Loads (E_{ss})

These are the loads generated by the Safe Shutdown Earthquake, which is the maximum potential earthquake that could occur in the vicinity of the site, based on geological and historical investigations. Dead and fixed equipment loads are described under the Operating Basis Earthquake, above. The horizontal and vertical forces on the containment are developed from the response spectra given in Figure 2.5-38 and Figure 2.5-39, the development of which is described in Subsection 2.5.2.6.

The effects of two (2) orthogonal horizontal earthquakes and one (1) vertical earthquake are considered and combined by the square-root-of-the-sum-of-the-squares method. Due to the presence of the containment enclosure, there are no dynamic effects of soil on the containment.

(b) Tornado Loads (W_t)

Due to the presence of the Containment Enclosure Building, wind pressure and pressure variation are not considered in the containment design. In addition, the containment structure is 3'-6" thick at the minimum, and tornado-generated missiles are not deemed capable of penetrating a 24" thick reinforced concrete wall (see Subsection 3.5.1.4). Therefore, missile effects due to the tornado are also not considered in the containment design.

5. Abnormal Loads

Abnormal loads are those generated by postulated high energy ruptures, particularly a rupture in the Reactor Coolant System resulting in a loss-of-coolant accident (LOCA). Post LOCA containment flooding is also considered. The maximum level of flooding, however, is 5'-4" above the top of the fill mat (-26 feet); this depth of water causes negligible loading on the containment structure.

(a) Accident Pressure (P_a)

A transient pressure load is used for the design of the containment. The maximum calculated internal pressure associated with the DBA is 49.6 psig. This provides a margin of 4.8 percent for the design pressure which is 52.0 psig. The pressure-transient curve for the containment is shown on Figure 3.8-9.

(b) Accident Temperature (T_a)

The transient temperature increase of the liner was considered in the design of the containment. The maximum liner temperature is 268°F. However, a maximum liner temperature of 271°F has been used in the design. The temperature transient curves for the containment liner are shown on Figure 3.8-10 and Figure 3.8-11.

The time-dependent thermal gradient through the concrete of the dome, cylinder and base mat was also considered in the design of the containment. When the accident pressure, P_a, is considered, the coincident thermal gradient is equivalent to the normal operating gradient. Due to the high insulating properties of the concrete, the pressure peak occurs before the temperatures within the concrete are appreciably altered. For design of the cylinder and dome, the peak liner temperature and peak pressure were also considered to occur simultaneously. This produced the most conservative design condition where responses to the loads are additive. Where responses are not additive, peak pressure was considered without the thermal loads. The thermal gradients used for the design of the containment are shown on Figure 3.8-8.

For the design of the liner, the transient conditions of liner temperature and coincident accident pressure were considered in order to produce the most conservative design.

(c) Accident Pipe Reactions (R_a)

Pipe reaction loads due to thermal conditions generated by the DBA, including R_o, were considered in the design. The magnitude of these loads was determined by the piping design and is included in the Design Report.

(d) Pipe Break Loads (R_r)

These are local effects on the containment due to the DBA, as follows:

(1) R_{rr} = load on the containment generated by the reaction of a ruptured high energy pipe. The time dependent nature of the load and the ability of the containment to deform beyond yield are considered in establishing the structural capacity necessary to resist the effects of R_{rr}.

(2) R_{rj} = load on the containment generated by jet impingement from a ruptured high energy pipe. In general, direct impingement of steam on the containment does not produce significant design loadings due to the distance between the wall and the break location. Where a break is postulated to occur close enough to produce a critical loading, a shield or deflector is provided, and the loading is transferred to the embedment for the pipe whip restraint to which the shield is attached.

(3) R_{rm} = load on the containment resulting from the impact of a ruptured high energy pipe. Generally, all high energy lines are constrained by pipe restraints, and loading of this nature is prevented. However, in isolated cases where there are postulated pipe whip impact loads, liner adequacy was evaluated for such loads.

6. Site-Related Loads

Site-related loads are loads peculiar to the Seabrook site. These loads are not combined with abnormal loads but are considered on an individual basis.

(a) Aircraft Impact

The impact of an FB-111 type aircraft weighing 81,800 pounds and traveling at 200 mph was evaluated. The effects of this aircraft crash on the containment have been investigated and were found not to be a controlling design consideration. For a complete discussion of the aircraft impact analysis, see Appendix 2P, entitled "Seabrook Station Containment Aircraft Impact Analysis."

(b) Turbine Missiles

The probability that a turbine missile will strike the containment has been shown to be less than 1.0×10^{-7} per year. Consequently, this type of missile was not considered in the design. See Subsection 3.5.1.3 for a complete discussion.

c) Explosions and Delayed Ignition of Vapor Cloud

Structural loadings from explosions and delayed ignition of a vapor cloud were shown to be of a very low magnitude and were not considered in the design. See Subsection 2.2.3.1 for a complete discussion.

(d) Flood Loads (F)

The effect of the design basis flood to Elevation +20.6' (50.6' above the top of the base mat) is considered in the design of the containment. Due to the presence of the Containment Enclosure Building, the only effect of the flood is its buoyant effect on the containment of the structure.

7. Other Pipe Break Loads

All high energy pipe lines are postulated to break, and all moderate energy lines are postulated to crack. The loads produced by these accidents, however, have no effect on the containment design, either because of shielding around the pipes or because of the distance separating the pipes and the containment wall.

8. Cyclic Loading

The various cycles loads were considered in the design. The following design conditions were considered in the fatigue analysis:

120 cycles start and shutdown

500 OBE cycles

100 SSE cycles

1 accident cycle (LOCA)

160 pressure test cycles (equipment hatch and personnel air locks)

b. Load Combinations

Various combinations of loads were used to determine the maximum strength required of the containment at various locations. These combinations were divided into service and factored load combinations.

1. Service Load Combinations

Service load combinations include conditions encountered during testing, normal operation, shutdown and severe environmental conditions; these are listed in Table 3.8-1.

Under these conditions, the structural components are designed to remain within elastic limits and satisfy the stress limitations specified in Subsection 3.8.1.5.

2. Factored Load Combinations

Factored load combinations include those conditions resulting from severe environmental, extreme environmental, abnormal, abnormal/severe environmental and abnormal/extreme environmental loads, as defined in Division 2 and listed in Table 3.8-1.

For each of these loading categories the structure is designed so that the allowable stresses comply with Article CC-3000 of Division 2, and the overall structural behavior is predicted to remain in the elastic range when thermal effects are not included. Design assumptions are presented in Subsection 3.8.1.4.

3. Additional Notes on Load Combinations

The load combinations, with their appropriate load factors, which require investigation to assure that the maximum effects of all load combinations are considered, are in accordance with the requirements of Division 2 and are given in Table 3.8-1. These combinations are used in the overall design of the concrete containment and also in the design of localized areas, such as penetrations and shell discontinuities.

For loads that vary, those values which produce the most critical combination of loading were considered. The live load (L) is considered to vary from zero to full value for all load combinations.

Long-term conditions, such as operating thermal loads, creep and shrinkage, which produce compression in the reinforcing steel, do not have a significant effect on the structural integrity of the containment structure, since the accident loads, which are the most significant loads, are generally resisted in tension by the reinforcing steel. In addition, the accident loads are short once-occurring loads which will have negligible creep effects. The self-straining loads associated with ASR expansion are considered for localized areas affected by ASR in combination with other loads as indicated in Table 3.8-1.

For the design of the liner, the load combinations in Table 3.8-1 are applicable with the exception that coefficients for all load cases are taken equal to 1.0.

Steel components of the containment shell that are pressure-resisting but unbacked by concrete are designed in accordance with the ASME Code, Section III, Division 1, using loads described in Subsection 3.8.2.

Earthquake effects are not assumed to occur simultaneously with flooding effects since the maximum flood is not associated with an earthquake.

Effects of a thermal gradient through the concrete section are not considered where the effects of the gradient reduce the effects caused by an abnormal loading condition.

Maximum values of time-dependent loads such as accident pressure, temperature and pipe break loads are considered.

The load combinations in Table 3.8-1 are applicable to the computations of factors of safety against overturning, sliding and flotation, with the exception that the coefficient for live load is zero. Buoyant forces are conservatively considered to decrease the dead loads for determination of overturning and sliding.

3.8.1.4 Design and Analysis Procedures

The containment structure is designed as a reinforced concrete thin shell structure in accordance with the requirements of Article CC-3000 of Division 2, as described in this Subsection and in other subsections of Subsection 3.8.1, and in accordance with the other applicable codes, standards and specifications defined in Subsection 3.8.1.2. The containment structure is designed to safely withstand the load combinations as defined in Subsection 3.8.1.3, and to provide biological shielding for normal and accident conditions. The critical areas for analysis were the base mat, the intersection between cylinder wall and base mat, the liner plate system, and the penetration openings.

The walls, dome and base mat of the containment are a reinforced concrete system with a leaktight steel liner attached to the inside surface, and were designed to behave as a single integrated system under the applied load combinations.

The containment structure as a whole behaves as a membrane structure, except in areas of discontinuity where there are local shear forces and moments. When subjected to internal pressure, the dome and walls displace outward with slight discontinuity effects at the dome and wall intersection. Discontinuity effects also exist near openings and at the intersection of the wall and base mat.

An iterative approach to design was taken, in which a proposed structural system was analyzed for stresses, strains and displacements using the finite element method, and then checked against design acceptance criteria, as defined in Subsection 3.8.1.5. The iteration process was repeated until an acceptable design was achieved.

The objective of the analysis of the reinforced concrete portion of the containment was the determination of maximum stresses in concrete and reinforcement over the range of boundary conditions, cracking assumptions and load combinations. To this end, the reinforced concrete was analyzed as a layered system, which allowed the investigation of each discrete layer of meridional, hoop and seismic reinforcing.

a. Design

The containment was designed to retain its functional capability during normal operation and emergency conditions. To meet this criterion, the leaktight integrity of the liner is maintained, and the structure was designed to respond elastically under all mechanical loading, except as noted in Subsection 3.8.1.5. The design criteria are based on the applicable codes discussed in Subsection 3.8.1.2.

Typical reinforcing details are shown on Figure 3.8-1, Figure 3.8-2, Figure 3.8-3 and Figure 3.8-4, Figure 3.8-6 and Figure 3.8-7.

b. Analytical Techniques

The analysis of the containment was performed by computer programs which utilize the finite element method. The finite element method was implemented by idealizing the containment with a system (model) of appropriately shaped elements which are interconnected at node points. These node points are located at the intersections of the lines which define the boundaries of each element. Generally, two types of analyses, axisymmetric and three-dimensional, were performed.

A finite element axisymmetric analysis, using solid and shell ring elements with orthotropic and elastic material properties, was used in the determination of displacements and forces in the containment structure with temperature, pressure and dead loads. The concrete, steel reinforcement, and the steel liner were represented by a system of ring elements which are triangular, linear, or quadrilateral in shape, as shown in Figure 3.8-12. The hoop, meridional and seismic reinforcement was modeled with shell elements located near the inside and outside surfaces of the wall and dome, as shown in Figure 3.8-13. Each layer of reinforcement was idealized as an independent uncoupled orthotropic steel plate.

Three-dimensional finite element analyses using elements of appropriately shaped sections were applied in the determination of displacements, stresses and strains for nonaxisymmetric loadings and local regions of the containment. Various types of elements were used for this analysis, such as shells and solids with isotropic and orthotropic material properties. The concrete, reinforcing steel and steel liner were idealized using layered elements. The three-dimensional finite element method was applied to the analysis of localized areas of the containment. A detailed discussion of the finite element programs used in the above mentioned analyses is given in Subsection 3.8.1.4g, and Appendix 3F.

c. Assumptions on Boundary Conditions

1. Containment Shell Analyses

The base mat was analytically treated as an integral part of the containment to realistically represent its effect on the containment shell behavior. The bottom of the base mat was assumed fixed in the vertical direction but free in the radial direction in all axisymmetric analyses for pressure, temperature and dead loads. For the seismic analyses described in Subsection 3.7.2, full fixity was assumed at the base of the containment wall.

Table 3.8-4 summarizes the load case and boundary condition assumptions for the containment shell analyses.

The containment mat with the reactor cavity pit and a portion of the shell was modeled in separate analyses. A discussion of boundary conditions is included in Subsection 3.8.1.41.

2. Large Opening Analyses

A quarter sector of the containment was modeled. This model extended from the base to the apex of the dome in the meridional directions and from a plane which passes through the opening to a plane at 90° to the centerline of the opening, as shown in Figure 3.8-14. The base was assumed fully fixed; the boundary conditions on the two sides of the model are defined as shown in Figure 3.8-14 and Table 3.8-5. These displacement boundary conditions correspond to the displacement degrees-of-freedom of the thick shell element discussed in Subsection 3.8.1.4j.

3. Aircraft Impact Analysis

The aircraft impact analysis described in Appendix 2P included two elastic dynamic analyses of the containment: an axisymmetric analysis (impact at apex of dome) and an asymmetric analysis (impact at springline). A shell model of the containment was used and full fixity was assumed at the base.

d. Axisymmetric and Nonaxisymmetric Loads

1. The primary behavior of the containment under axisymmetric loads is predominantly thin shell, membrane behavior except in regions near the base, springline and large openings where discontinuity moments and shears are significant. These latter effects are of localized nature and are discussed further in Subsections 3.8.1.4e and 3.8.1.4j. Thermal, pressure and dead loads were applied as axisymmetric loads to the axisymmetric containment model. These are the first four load cases listed in Table 3.8-4. The cracking patterns tabulated in the second column of the table were found by an iterative method. This is discussed further in Subsection 3.8.1.4f. The analyses were performed using the finite element computer code, Wilson I, "A Finite Element Analysis of Axisymmetric Solids Subjected to Symmetric Loads," by E. L. Wilson. This code is described in Subsection 3.8.1.4g and in Appendix 3F.

Figure 3.8-13 illustrates the reinforced concrete idealization used in the model, and Figure 3.8-15 shows the shell-type force and moment resultants which are calculated for sections along the shell meridian, and subsequently used in the design of the reinforcement. Figure 3.8-16 illustrates the axisymmetric containment model with integral mat, wall and dome used in the first four analyses of Table 3.8-4. Representative node points and section numbers are shown where the latter range from 1 at the apex of the dome to 98 at the base of the wall. At any section, the model is composed of shell and solid elements representing the steel liner and reinforcement and concrete, respectively. Material properties are assigned to each layer according to the orientation of reinforcement and cracked or uncracked state of concrete. Force and moment resultants were computed for each load case by integrating stresses through the cross section. These cross-sectional forces and moments are combined for all applicable load combinations of Subsection 3.8.1.3.

2. <u>Nonaxisymmetric Loads</u>

Nonaxisymmetric loads applied to the containment structure, other than localized loads, such as internal missiles, are the seismic and aircraft impact loads. The seismic input and seismic analysis are discussed in Subsections 3.7.1 and 3.7.2, respectively. The aircraft impact load and analysis are described in Appendix 2P. Wind and tornado loads are not considered because of the presence of the Enclosure Building. An elastic analysis of the aircraft impact loading was made using the Wilson II finite element code, "Dynamic Stress Analysis of Axisymmetric Structures Under Arbitrary Loading," by Ghosh and Wilson, as revised in September 1975. This code is further described in Subsection 3.8.1.4g and Appendix 3F. The loading was represented by Fourier series functions applied to the axisymmetric model of the containment.

The shear forces and overturning moments due to OBE and SSE seismic loads were obtained from the seismic analysis described in Subsection 3.7.2. These shear forces and moments were treated as effective static forces acting over the cross-section of the containment shell. Modeling assumptions for these two load cases are shown in Table 3.8-4.

3. Design of Reinforcement

The force and moment resultants for the individual load cases were combined for the load combinations of Subsection 3.8.1.3 and Article CC-3000 of the ASME Code. The provisions of Subarticle CC-3432 of Division 2 were used in the sizing of reinforcement steel.

e. Transient and Localized Loads

A "hot liner" transient load was analyzed and accounted for in the design of the reinforced concrete wall. A temperature spike was placed on the liner for two cases: (a) "hot liner" with no temperature gradient on concrete wall, and (b) "hot liner" with a temperature gradient through the wall (see Table 3.8-4). When combined with mechanical loads, the effect of thermal loadings is a redistribution of stresses and strains on the cross section. The stresses and strains in the liner are discussed in Subsection 3.8.1.4k, "Steel Liner Plate and Anchors."

The aircraft impact analysis is described in detail in Appendix 2P. Three analyses were performed: two of the overall structural behavior and one localized elasto-plastic dynamic analysis. The analyses of overall behavior considered the conditions of impact on the dome (axisymmetric structure with axisymmetric loading) and impact on the springline (axisymmetric structure with unsymmetric loading). Both analyses assumed linear behavior and used the Wilson II finite element code. The asymmetric loading of the second analysis was represented by a Fourier series. Both analyses showed that yielding would occur local to the point of impact. Accordingly, a localized, nonlinear analysis was used to determine the extent of damage to the containment shell. The details of this analysis are also found in Appendix 2P. In brief, effective mass and stiffness properties were determined for an assumed mode of collapse consisting of a circular fan yield-line configuration.

An equivalent single-degree-of-freedom nonlinear model was then subjected to an idealized force-time loading function and the maximum deformations determined. It was shown that the "as designed" containment structure with Enclosure Building can withstand impact of an FB-111 aircraft at 200 -mph impact speed without collapse or impairment of the leak-tight integrity of the liner.

The localized strain from ASR expansion is determined from measurements of the structure in ASR-affected locations. Closed form analytical equations for cylindrical shells subjected to local loadings were used to calculate the forces and moments to localized ASR expansion. The ASR loads are combined with other loads for the structure to analyze critical areas affected by ASR.

f. Creep, Shrinkage and Cracking of Concrete

Due to low sustained concrete stress associated with conventionally reinforced concrete structures, the effects of concrete creep are negligible. Since the load combinations which control the rebar design involve accident pressure, which effectively cracks the concrete and places the reinforcement into tension, creep and shrinkage-induced stresses are not limiting factors in design. In addition, the structural integrity test cracks the concrete thereby relieving creep and shrinkage stresses that occur subsequent to construction. Therefore, the effects of creep and shrinkage were not considered in the analysis and design.

Since it was assumed that concrete has no tensile capacity, a cracked section, iterative approach was used for the analysis under the thermal, pressure and dead loads. The procedure for determining the crack pattern involved a check of the stresses in each concrete layer of the cross section. Where concrete stresses were in tension, the elastic properties in that direction (modulus of elasticity and associated Poisson's ratio) were set equal to zero. The procedure was repeated until there were no significant changes in moments and forces on the containment sections in two successive iterations. Since the finite element representation of the cross-section modeled both concrete and rebar using a layered model, as shown in Figure 3.8-13, an accurate portrayal of cross section stiffnesses under various configurations of cracking was possible. Table 3.8-4 defines the cracking pattern used for each load case. The analysis of the containment subjected to the nonaxisymmetric seismic loads considered the concrete to be uncracked. For this load case, the magnitudes of flexural moments are small compared to the flexural moments resulting from the axisymmetric loads. Hence, the redistribution of moments due to cracking under seismic loading does not control the design.

g. Description of the Computer Programs Utilized in the Design and Analyses

The computer programs used in the containment design and analysis are briefly described in this subsection. The programs are either of two types: a recognized program in the public domain with sufficient history of use and documentation to justify its applicability and validity without further demonstration, or a program which gives solutions to a series of test problems that have been demonstrated to be substantially identical to those obtained from classical solutions and/or analytical results published in technical literature. Utility programs used to replace hand calculations are not discussed. These programs were validated by comparison to sample hand calculations whenever used in the analysis.

1. STARDYNE, Static and Dynamic Structural Analysis System, by Mechanics Research, Inc., 9841 Airport Boulevard, Los Angeles, CA., 90045. Documentation is available from Control Data Corporation (Publication No. 76079900). The STARDYNE system is designed to analyze linear elastic structural models for a wide range of static and dynamic problems.

2. MARC-CDC, Nonlinear Finite Element Analysis Program, by Dr. Pedro Marcel and Associates of the MARC Analysis Corporation, 260 Sheridan Ave., Palo Alto, CA., 94306. Documentation is available from Control Data Corporation. MARC-CDC provides elastic, elastic-plastic, creep, large displacement, buckling and heat transfer analysis capabilities. It also performs dynamic analysis by the modal or direct integration procedures.

3. WILSON 1, (SAG 001) Finite Element Analysis of Axisymmetric Solids Subjected to Axisymmetric Loads, by E. L. Wilson of the University of California, Berkeley, July 1967-Revised, November 1969. Documentation is available from the Earthquake Engineering Research Center of the University of California, Berkeley. The Wilson 1 computer program is based on the finite element direct stiffness method, and is applied to the determination of stresses and displacements in axisymmetric structures (solids and/or shells of revolution) subjected to axisymmetric mechanical loads or temperature gradients. The theoretical basis of the program is the work of E. L. Wilson, References 1 and 2.

4. WILSON 2, (SAG 010), Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loading, by S. Ghosh and E. L. Wilson of the University of California, Berkeley, September, 1969-Revised September 1975. Documentation is available from the Earthquake Engineering Research Center of the University of California, Berkeley, Report, No. EERC 69-10. The Wilson 2 program is based on the finite element stiffness method and is applied to complex axisymmetric structures subjected to any arbitrary static or dynamic loading or base excitation. The three-dimensional axisymmetric continuum is represented as an axisymmetric thin shell, a solid of revolution, or as a combination of both.

5. LESCAL calculates the stresses and strains in rebars and/or concrete in accordance with the criteria set forth in Subarticle CC-3511 of Division 2. The section may be reinforced with horizontal, vertical and diagonal rebars. The applied loads are axial forces and moments in the vertical and horizontal faces and in-plane shear. When in-plane shear forces are included, a solution is obtained by solving Duchon's equations, Reference 3.

6. SAG 054, Amplified Floor Response Envelope. This program generates an envelope for amplified response spectra, spreading the peaks by a user-specified amount.

7. SAG 058, Response Spectra. This program calculates the response spectra of a single degree-of-freedom damped oscillator due to a transient base motion. The input base motion may be an arbitrary forcing function. The output consists of the maximum relative displacement, the maximum relative velocity and the maximum absolute acceleration for the various selected frequencies and the times when these values occur.

8. TAPAS, (SAG 008), "Transient Temperature Analysis of Plane and Axisymmetric Solids," Reference 4, was developed to determine the temperature distribution through a solid body as a function of time when subjected to temperature variation or heat flux inputs. A finite element technique coupled with a step-by-step time integration procedure is used. Both steady-state and transient heat flow can be treated.

9. SAG 017, Fourier Coefficient Expansion Program, was developed to be used in conjunction with the Wilson 2 program to compute Fourier series representation of general nonaxisymmetric load functions.

10. SAG 024, MMIC, calculates weight, weight moments of inertia and plan location of the center of weight of a segment of a structure given the dimensions, density and location of each structural component and the magnitude and location of all concentrated loads.

11. SAG 025, SECTION, calculates beam section properties of structures for use in lumped mass stick models for dynamic analysis.

12. ANSYS, Finite Element Analysis Program, by ANSYS, Inc., Canonsburg, PA, USA. ANSYS is a general purpose finite element commercial software. It provides broad range of analysis capabilities including static and dynamic analyses. This program is used for ASR deformation analysis.

h. <u>Tangential Shear</u>

The design of the containment for seismic loads incorporates diagonal shear reinforcing to resist tangential shear. The design for tangential shear is in accordance with the provisions of Article CC-3000 of the ASME Code, with the exception of the limitation on allowable concrete shear stress shown as Note 1 of Table 3.8-2.

The forces and moments due to seismic loading are discussed in Subsection 3.7.2. Three components of the seismic motion were assumed to exist simultaneously, and the resulting component forces and moments were combined by the square-root-of-the-sum-of-the squares (SRSS) method. The maximum tangential shear from the SRSS combination is assumed to act simultaneously at all points on the containment circumference at the given elevation. All forces and moments were combined per the specified load combinations. The LESCAL program, described in Subsection 3.8.1.4g, was used to calculate rebar stresses for all sections and elevations combining the stresses due to in-plane forces and moments. Duchon's equations (Reference 3) are incorporated into LESCAL for calculating rebar stresses (including inclined rebar) for the combined membrane forces.

i. <u>Variation in Physical Material Properties</u>

The effects of variations in material properties were considered in the design and analysis. Material properties which can strongly influence both analysis and design due to variability or uncertainty include: (1) dynamic modulus of soils, (2) the modulus of elasticity of concrete and, (3) material strengths.

As this containment is founded on rock, the first of these sources of variability is removed from consideration. The modulus of elasticity of concrete is a function of concrete compressive strength which in turn is typically substantially higher in the "as-built" structure than assumed for analysis and design. While variability in concrete modulus has no significant effect on structural design, it influences structural stiffness and natural frequency, and, subsequently, the amplified response spectra of the seismic analysis. This impacts equipment design as discussed in Subsection 3.7.3. The variability was accounted for by peak spreading when generating envelopes of the response spectra. Variability in material strength is taken into account in Division 2, Subarticle CC-3400, design allowables.

Analyses and tests were completed to assess the effects of ASR on structural properties of reinforced concrete. The results indicate that using the structural properties and code equations from the original design analysis is conservative when ASR expansion levels are below the limits defined in Subsection 3.8.4.7. Concrete expansion from ASR imposes a localized tensile strain in the reinforcement.

j. Thickened Penetrations

The equipment hatch and personnel air lock (27'-5" and 7'-0" diameters, respectively) are large openings with thickened bosses. Figure 3.8-3 and Figure 3.8-4 are drawings of typical wall sections with their reinforcement configurations. The wall thicknesses and reinforcement were designed for the stress concentration effects induced by the presence of the openings in the cylinder. The main hoop and meridional reinforcement is bent around the opening to provide continuity. Additional local reinforcement is provided, including stirrups and tiebacks for bent bars.

The analyses of the large openings were by the MARC finite element computer code using a three-dimensional model of a quadrant of the containment cylinder and dome, Figure 3.8-17. The two openings are sufficiently separated to allow independent analyses. Therefore, two planes of symmetry were assumed and the single quadrant model resulted. Appropriate symmetry and antisymmetry in loads and boundary conditions were defined. Figure 3.8-14 and Table 3.8-5 show the boundary conditions for the various load cases. The model extends from the base to the apex of the dome using a thick-shell superparametric element (element 22 in the MARC program element library). This element incorporates eleven through-thickness layers to accurately represent the liner, reinforcement and concrete. Figure 3.8-18 illustrates the partitioning of a typical cross-section into the 11 layers. The tapered-thickness transition region between the normal 54-inch wall and the thick boss is modeled by stepping the element thicknesses through this region. The thick-shell element gives all stresses, except the through-thickness (normal to wall) direct stress, at each of the eleven layers. The conventional shell-type force and moment resultants, including transverse shears, are calculated from these stresses and were used as the basis for design of the reinforcement.

Design of the reinforcement was accomplished in conformance with Division 2 provisions for load combinations and material stress allowables, as described in Subsections 3.8.1.3 and 3.8.1.5, respectively.

k. Steel Liner Plate and Anchors

1. Design

The liner is anchored to the reinforced concrete with embedded stiffeners and anchors. Typical liner details are shown in Figure 3.8-5. To maintain the leak-tight integrity of the liner under service loads and factored loads, the liner was designed to follow the major strain pattern of the surrounding concrete. The liner plate was thickened in areas around penetrations as required. The stresses and strains in the liner, determined by the analysis, are within the design allowables given in Table CC-3720-1 of Division 2.

The anchorage system was designed so that it can accommodate the in-plane loads or deformations exerted by the liner plate and/or loads applied normal to the liner surface. The anchorage system was designed so that a progressive failure of the anchorage system is precluded in the event of a defective or missing anchor. The design of the anchors considered the effects as indicated in Subarticle CC-3810 of Division 2.

The displacements and forces in the anchorage system obtained by analysis are not allowed to exceed the allowables given in Table CC-3730-1 of Division 2.

Penetrations are provided with an anchorage system capable of transferring thermal loads, pressure loads and other mechanical loads, such as piping reactions, to the concrete containment. The penetrations, brackets and attachments are designed in accordance with the provisions of the ASME Code, Section III, Division 2. Designing those portions of the penetrations which fall within the jurisdiction of Division 1, that is, those portions not backed by concrete, is described in Subsection 3.8.2.

2. Analysis

The stresses and strains in the liner due to dead load, internal pressure, a "hot" liner and a temperature gradient were obtained from the containment analyses using an axisymmetric finite element model, as discussed in Subsection 3.8.1.4d. These analyses considered the liner to be integral with the concrete structure. The liner anchors were analyzed assuming the unbuckled liner remained elastic under all conditions. Construction and material imperfections described in Division 2, Subarticle CC-3810, were considered in the liner anchor analysis including the possibility of a buckled liner panel. Liner stresses were assumed for the combined load cases. These stresses produced liner/anchor displacements in the direction of the buckled panel. An analytical model was developed in which the buckled panel was assumed to be elastic-perfectly plastic and was represented by an equivalent non-linear spring whose stiffness was determined by a separate nonlinear analysis using the MARC-CDC finite element code (Subsection 3.8.1.4g). The anchors were also represented by equivalent nonlinear springs whose force-displacement properties were determined by testing. The analytical model yielded forces and displacements of all anchors.

l. Containment Mat

The containment mat, the reactor cavity pit and 55' of the cylindrical containment shell above the mat were modeled with Element 22 in the MARC-CDC finite element program. A portion of containment shell in addition to the reactor cavity pit was included in the model to represent the stiffening effect of the containment shell on the mat. The internal structure and the fill mat were not modeled, but the loads which they transmit to the structural mat were considered. The stiffening effect of the dome and the cylindrical portion of the shell above the top of the model was considered by providing additional constraint equations which enforce the plane section remaining plane assumption. In effect, these equations insure that the top of the model remains circular and lies in a plane after deformation.

The rock foundation supporting the containment structure was modeled using the continuous foundation option which is available in the MARC-CDC program. This option requires as input foundation properties in both tension and compression at 9 integration points on the surface of each foundation element. To account for lift off, a zero stiffness was specified in tension. A very large stiffness is given to represent the rigid behavior of the rock foundation in compression. There were no vertical boundary conditions prescribed at node points on the mat. The consistent formulation in the element automatically accounts for this vertical restraint in compression. The stiffness of the internals and fill mat was not considered since there is no structural connection between these structures and the structural mat. Because of symmetry, only one-half of the structure was modeled, (Figure 3.8-19).

The following boundary conditions were applied to the model:

1. All node points in the global 134 xz plane have symmetry boundary conditions as given below:

   $u_y = \theta_x = \theta_z = 0$

   where:

   u_y = translation in the global y-direction

   θ_x = rotation about the global x-axis

   θ_z = rotation about the global z-axis

The rotational boundary conditions are input into the program in a local coordinate system.

2. All node points at the outside edge of the mat within + 50° from the direction of the global x-axes have been restrained in the global x-direction to account for the lateral restraint provided by the rock below grade. (The earthquake forces are assumed in global x-direction.) Thus, all horizontal loads are resisted in bearing through the reactor cavity pit and a circular arc at the elevation of the centerline of the mat. The boundary conditions used in the mat analysis differ from those used in the axisymmetric analyses since the mat is represented by a three-dimensional model rather than the two-dimensional axisymmetric model. This permitted a more accurate representation of the boundary effects. The mat was modeled in the axisymmetric analyses only to simulate the mat restraining effect on the containment shell.

The mat was analyzed for dead, live, pressure and seismic loads. Seismic loads were applied as inertia forces which were calculated from absolute accelerations computed in the seismic analysis, Subsection 3.7.2. At the top of the model, external forces were applied which represent the internal forces at that elevation for each load combination. Dead load, pressure, and vertical seismic loads were assumed uniformly distributed, and the moments and shears from the horizontal earthquake were distributed as normal and shear forces which vary as cosine and sine functions, respectively.

The nonlinear effects resulting from uplift required an iterative analysis. Five iterations were required to reach the equilibrium state.

When combining seismic loads with other loads, due to the inherent nonlinearity associated with mat uplift, the response due to the vertical component of the earthquake has been added algebraically to that of one horizontal component. Since the final stress state in a nonlinear problem can only be determined when all loads are applied simultaneously, it is not possible to separate out the individual contributions to the total response of two horizontal components and a vertical component of an earthquake in order to perform the SRSS. Consequently, only one horizontal component and the vertical component were applied and no SRSS process was used. The horizontal component was assumed to act along the longer direction of the reactor cavity pit (x-direction).

m. An ultimate capacity analysis of the containment structure for internal pressure loads was performed. The pressure-retaining capacity of the overall containment structure and the localized areas was determined. The analysis was based on actual material properties.

n. A Design Report (9763.102-CDR-1) of the containment structure was prepared in accordance with the requirements of Division 2. This report contains sufficient information to substantiate that the design of the containment structure is in accordance with the requirements of the containment design specification and the ASME codes.

3.8.1.5 Structural Acceptance Criteria

a. General

The containment structure, including liner and penetrations, was designed to remain within elastic limits under service load conditions and under the mechanical loads of the factored load conditions. With thermal loads included, the reinforcing steel yielded in some regions but was within the allowable strain limit of $2 \times \in_y$. Gross deformations of the containment were also checked to assure that there is no interaction with other structures or components.

The design limits imposed on the various parameters that serve to quantify the structural behavior and provide a margin of safety are in compliance with Article CC-3000 of Division 2. The allowable limits on these parameters, for service and factored loads, are given in Table 3.8-2.

b. <u>Concrete</u>

The allowable compressive stresses, including membrane, membrane plus bending and localized stresses, and shear stresses under service loads and factored loads are as specified in Article CC-3400 of Division 2, with the following exceptions:

1. The allowable shear stress, v_c, to be resisted by the concrete will not exceed 40 psi and 60 psi for load combinations 7 and 8, respectively, in Table 3.8-1.

2. The allowable concrete stresses or radial shear are based on Code Case No. N-287.

3. The allowable concrete stresses for peripheral shear are based on Code Case No. N-219.

c. <u>Reinforcing Steel</u>

The stress and strain limitations for reinforcing steel, under service loads and factored loads, are as specified in Subarticles CC-3432 and CC-3422 of Division 2, respectively.

Stress concentrations normally occurring around openings and penetrations were controlled by the use of additional reinforcing steel, which resulted in stresses and strains within the above indicated limits.

There is, however, some yielding of the seismic (diagonal) reinforcing under the mechanical loads of the abnormal/severe environmental and abnormal/extreme environmental loading conditions at the edges of the transition regions below the equipment hatch and personnel air lock adjacent to what may be considered the membrane region. The net strain at these locations, with thermal load included, is less than $2\in_y$ (.00414 in/in), which is the limit established in Subarticle CC-3422.1 (d) of Division 2 (1977 Winter Addendum). The structural integrity of the containment is not affected, however. The strain limit of $2\in_y$ insures that the yielding under thermal load does not result in concrete cracking which would cause deterioration of the containment.

d. Liner Plate and Liner Anchorage System

Tensile and compressive stress/strain limits in the liner plate, including membrane and membrane plus bending conditions, are in accordance with Subarticle CC-3700 of Division 2. (For the liner plate, the 1976 Winter Addendum of Division 2 is used, and for anchorage system, the January 1, 1975 Edition of Division 2 is used. All supporting documentation for material procurement and fabrication activities that commenced prior to July 1, 1975, included Certificates of Conformance to the technical requirement of this edition.)

The allowable values for forces and displacements of liner anchors embedded in concrete are based on test data, and are in accordance with the limits of Table CC-3730-1 of Division 2. Tests were performed to obtain the shear load - displacement relationship of liner anchors. The details of liner anchor load test procedure and results are included in Appendix 3G.

The stresses and strains in penetration assemblies, brackets and attachments are also in accordance with allowables given in Subarticle CC-3700 of Division 2.

e. Stability

Acceptance criteria for stability against overturning, sliding and flotation are given in Subsection 3.8.5.5.

3.8.1.6 Materials, Quality Control and Special Construction Techniques

Materials used for the containment include concrete, reinforcing steel, liner plate steel and attachments thereto, and coatings, the requirements of which are in compliance with Article CC-2000 of Division 2 and the applicable NRC Regulatory Guides listed in Subsection 3.8.1.2.

Quality control procedures employed for the fabrication and construction of the containment are in compliance with Articles CC-4000 and CC-5000 of Division 2 and with the applicable NRC Regulatory Guides listed in Subsection 3.8.1.2. For all materials, suppliers were required to furnish Certified Materials Test Reports (CMTR), prepared in accordance with the requirements of Subarticle CC-2130 of Division 2. A CMTR includes results of all required chemical analysis, physical tests, mechanical tests, examinations including radiographic film, repairs and heat treatments performed on the material.

Materials used in construction, as well as their respective quality control procedures, are further described in the sections that follow. Engineering properties are given in Table 3.8-3. Construction tolerances are in accordance with the criteria of Division 2.

There are no special construction techniques.

a. Concrete

The ready-mixed concrete which was used is a dense durable mixture of sound coarse aggregate, fine aggregate, cement, water and admixtures, in accordance with the material, proportioning, mixing, transporting, placing and testing requirements of Division 2.

Testing which was performed on the concrete and on its components is described in UE&C Specification 9763.006-69-1, "Concrete Batch Plant." Details of the placing procedures, including hot and cold weather precautions, are described in UE&C Specification 9763.006-13-2, "Containment Concrete Work." In the paragraphs which follow, however, a brief description of the material is presented.

Except as noted below, the containment is constructed of concrete which has a standard compressive strength at 28 days of at least 3000 psi. The base mat, reactor pit, bottom 10 feet of the cylinder, and the regions in the cylinder near the equipment hatch and personnel air lock, in which the reinforcing anchor plates are located, are constructed of concrete which has a standard compressive strength at 28 days of at least 4000 psi. To insure that these strengths were attained, verification testing was performed in accordance with the requirements of Division 2.

The nominal density of the reinforced concrete was considered as 150 lbs/ft^3. The mix proportions were established on the basis of laboratory trial batches which were designed by the testing laboratory per ACI 211.1-74. The concrete was produced under controlled conditions by a fully automatic central batch plant located on the site to assure that the proper ingredients and proportions determined by the design mix were achieved.

The cement conforms to the requirements of ASTM C150, "Specification for Portland Cement," Type II, having a low alkali content and a moderate heat of hydration.

The coarse and fine aggregates conform to the requirements of ASTM C33, "Specification for Concrete Aggregates" and the additional requirements of Subarticles CC-2222 and CC-3421.5.1 of Division 2.

The maximum coarse aggregate size in the containment is 1½", except in congested areas where a 3/4" (size #7, as specified in ASTM C33) maximum aggregate size mix was used to accommodate proper placement of the concrete.

The mixing water conforms to the requirements of Subarticle CC-2223 of Division 2.

To assure a plastic and workable mix, increase durability, and increase ease of placement in congested areas, admixtures were used in the concrete mix design. These admixtures consisted of air entraining agents, water reducing agents, and retarding agents. Their effects on the strength of the mix were considered in the mix design, such that the properties described previously were the properties which were obtained after the inclusion of admixtures.

Admixtures containing chloride ions were not used in the concrete for the containment.

The maximum slump permitted in mass concrete for the containment was 3", except in congested areas where a 4" slump was allowed to accommodate proper placement, with slumps greater than 4" but not more than 6" (Special High Slump Concrete) used in highly congested areas. The maximum slump for concrete utilizing a superplasticizer (high range water reducer) was 8" (9" on a case-by-case basis). The maximum slump permitted for all other concrete was 4".

No aluminum materials were used in the mixing, handling, storing, transporting, or placing of concrete materials or mixes, nor were any aluminum embedments used.

The maximum concrete mix temperature during placement was 80°F.

All concrete operations during cold weather conditions followed the practice defined in ACI 301 and 306R-78 except that concrete as placed shall not be lower than 45°F. Concrete was maintained at 50°F.

During cold weather curing of the concrete, concrete surfaces whose temperatures are below 50°F by accident for short periods of time, but remain 40°F or above, have had the 7-day curing period extended by the amount of time the concrete was below 50°F (rounded out to the nearest whole day).

Alkali Silica Reaction is discussed in subsection 3.8.4.6.

b. Reinforcing Steel

Reinforcing steel consists of high-strength deformed billet steel bars conforming to ASTM A615, Grade 60. This steel has a minimum yield strength of 60,000 psi, a minimum tensile strength of 90,000 psi and a minimum elongation of 7 percent in an 8" gage length.

In addition to the Certified Material Test Reports, user tests, as required by Division 2 and Regulatory Guide 1.15, were performed by the Material Manufacturer on full-size diameter test specimens to further verify the physical properties of the rebar.

Arc welding of rebar was not permitted.

All reinforcing bars were detailed by the Manufacturer in accordance with the requirements of the Design Drawings. Detail drawings were reviewed by the Designer.

All reinforcing bars were spliced in accordance with Division 2 and UE&C Specification 9763-WS-4C. No. 14 and 18 bars were joined by mechanical butt splices (Cadweld splices). These splices met the requirements of Subarticle CC-4333.4.4. of Division 2. The splice sleeve material conformed to ASTM A519 (85 ksi - Minimum Ultimate Tensile Strength). Cadweld anchor plates used as mechanical anchorage for the terminated rebars, predominately in the equipment hatch and personnel air lock areas, conformed to SA537, Class 1.

The splices were tested in accordance with the requirements of UE&C Specification 9763-WS-4C to assure that they develop the specified strength. This test program is in accordance with the requirements of Division 2 and NRC Regulatory Guide 1.15, as discussed in Section 1.8, Regulatory Guide 1.136.

c. Containment Liner

The steel liner plate is carbon steel conforming to ASME SA 516, Grade 60. This steel has a minimum yield strength of 32,000 psi and a minimum tensile strength of 60,000 psi, with an elongation of 21 percent in an 8" gauge length, when tested to failure.

The materials, standards and quality control procedures used for the liner and penetrations are in accordance with Division 2 and NRC Regulatory Guide 1.19. Inspections are made in accordance with Article CC-5000 of Division 2.

The following materials were used for the penetration sleeves, equipment hatch and personnel locks:

1. Equipment Hatch-ASME SA-516, Gr. 60 Normalized
2. Personnel Locks-ASME SA-516, Gr. 70 Normalized
3. Fuel Transfer Sleeve-ASME SA-240, Type 304 Stainless Steel
4. Seamless Penetration Sleeves-ASME SA-333, Gr. 1
5. Rolled Penetration Sleeves-ASME SA-516, Gr. 60 Normalized.

Permanent attachments to the liner were fabricated from SA-36 or SA-516 Grade 60 or SA-516 Grade 70 plate. The anchorage system, tees, brackets and attachments to the liner plate were ASME SA-36.

All backing strips were of the same material specification as the item being welded.

All welding materials conformed to the requirements of UE&C Specification No. 9763-WS-4A and ASME Section II, Material Specification, Part C - Welding Rods, Electrodes and Filler Metals.

Studs and stud welding materials were in accordance with the requirements of Subsection CC-2620 of Division 2.

All liner materials are normalized, and sections in excess of 5/8" thickness were impact tested to 15 ft-lbs. at 20°F in accordance with the requirements of Division 2. For penetrations, the equipment hatch and personnel air lock, the test temperature was -10°F in the heat affected zone and -25°F in the parent metal.

When required, welds were post-weld heat treated in accordance with the requirements of Division 2.

The portions of the equipment hatch and personnel air lock within the jurisdiction of Division 1 were designed and detailed by the Fabricator. These portions are described in Subsection 3.8.2.

d. Steel and Concrete Coating System

The coatings used on steel, other than stainless steel, and concrete surfaces inside Containment, that might be exposed to spray have been tested and accepted in accordance with the requirements of ANSI Standard N101.2, except in the acceptance criteria used, and were applied in accordance with the manufacturer's printed instructions over properly prepared surfaces. The acceptance criterion for power tool cleaning methods, intended for limited use in the containment, is adherence of the coating (no solid debris generated) rather than specific ANSI blister size and frequency. (See also Section 6.1 for further discussion on BOP and NSSS equipment and structures coating systems inside Containment.)

Ferrous surfaces to which coatings were applied were abrasive blast cleaned in accordance with the Steel Structure Painting Council Specifications SP6 and SP10, except for the limited use of qualified power tool cleaning methods where blast cleaning became impractical. All horizontal and vertical concrete surfaces which were coated were washed and neutralized just prior to the initial coating application in order to produce a clean contamination-free surface. All concrete patching work, such as pointing of form tie holes and removal of sharp edges and fins, was completed before the washing was begun.

e. Grout

Sand Cement Grout used in general repair and patch work was in conformance with the requirements of ASME Section III, Division 2 Code, 1975 edition including Winter 1975 Addenda Subarticle CC-2240.

Prepackaged cement grout used in general repair and patch work met the requirements of ASME Section III, Division 2 Code, 1975 edition including Winter 1975 Addenda Subarticle CC-2240 except as modified herein. Aggregates met the requirements of ASTM C33 except that the gradations were adjusted as required to meet the Material Manufacturer's requirements for the product's applications.

The ASTM C88 Standard Test method for soundness of aggregates was not required to be performed provided alternate testing was performed by the Material Manufacturer to assure that the grout provided adequate resistance to weathering action.

The Material Manufacturer provided a notarized Certificate of Conformance (C of C) and Certified Material Test Reports (CMTRs) which contain the results of tests required in Corps of Engineers' CRD-C-589-70 and CRD-C588-78A. Compressive strength of grout was tested in accordance with CRD-C-588-78A, Section 10.3 "Compressive Strength" except that nine (9) cubes were made from each test mixture, and cubes were tested for compressive strength at 3, 7, and 28 days. CMTRs and Master Builders 713 Grout did not include chemical analysis.

f. <u>Epoxy Bonding Materials</u>

Sikadur High-Modulus Epoxy Bonding Adhesive (Sikastix 370) was used in general concrete repair and patching work when directed by the Construction Manager. The bonding adhesive was stored, mixed and applied in accordance with the Material Manufacturer's instructions.

The Material Manufacturer provided a notarized Certificate of Conformance for each batch of material supplied.

3.8.1.7 Testing and In-Service Inspection Requirements

The structural testing and in-service inspection program consists of the integrated leak rate test, in-service leak rate testing, the preoperational structural integrity test, and general visual inspection of structurally critical areas.

The in-service leak rate testing of the containment and visual inspection requirements are discussed in Subsection 6.2.6. The preoperational structural integrity test and visual examination are described below.

Since no new or previously untried design approaches are used for the containment structure, there are no special testing or in-service surveillance requirements.

a. <u>Structural Integrity Test</u>

To demonstrate that the concrete containment structure will respond satisfactorily to the postulated internal pressure loads, a preoperational structural integrity test (SIT) will be performed at 1.15 times the containment design pressure of 52 psig and at an average differential temperature between the inside and outside of the containment (within the Enclosure Building) not to exceed 65°F.

The SIT will be conducted in accordance with the nonprototype requirements of Article CC-6000 of Division 2. The Consolidated Edison's Indian Point Unit No. 2 and Washington Public Power Supply System Unit No. 1 are the prototypes for this test. Structural acceptance is based on gross deformations (diameter change, vertical growth and radial growth at the equipment hatch), concrete crack widths, deflection recovery and post test visual examination of concrete and liner. Prior to the test, a table of predicted gross deformations, crack widths, etc., will be provided as a guide for verifying satisfactory structural response during the test. These acceptance criteria will be as follows:

1. No yielding of reinforcement as determined by analysis of crack width and deflection data.

2. No visible signs of permanent damage to the structure or liner.

3. The measured maximum deflections at points of maximum predicted deflection shall not exceed predicted values by more than 30 percent. This requirement will be waived if the 24-hour recovery is greater than 80 percent.

4. The deflection recovery 24 hours after complete depressurization shall be a minimum of 70 percent.

   The instrumentation required to obtain the data needed to verify the structural response will include methods to measure radius and diameter deformation, vertical deformation, deformation around the large equipment hatch opening and breech-type personnel air lock opening, and crack measurements as required by ASME Section III, Division 2. The data will be obtained without embedding any devices in the cylinder or dome. A test report will be provided which will compare test results with predicted and allowable limits and evaluate any deviations.

   Before and after the test, the containment will be visually examined to assure that no distress has developed on the concrete or liner.

3.8.2 Steel Containment

The following sections contain the physical descriptions, applicable codes, standards and specifications, loads and load combinations, design and analysis procedures, allowable stresses, quality control and testing requirements for the major steel penetrations of the concrete containment structure that are pressure-resisting but unbacked by concrete. The containment structure itself is constructed of reinforced concrete and, as such, is discussed in Subsection 3.8.1.

3.8.2.1 Description of the Containment Penetrations

The containment penetrations described in this Subsection are the personnel air lock and equipment hatch, the fuel transfer tube assembly, the piping, electrical and instrumentation penetrations, and the ventilation penetrations. These components penetrate the containment shell to provide access, anchor piping, or furnish some other operational requirement. They also maintain leak tightness for the containment shell, and meet the maximum allowable leakage rate, as described in Subsection 3.8.1. Other functional characteristics are described below in the descriptions of individual components.

All penetrations are anchored to sleeves (or to barrels) which are embedded in the concrete containment wall. This embedment is accomplished by means of an engineered anchorage system that is welded to the sleeve (or barrel) which is, in turn, welded to the locally thickened liner (see Subsection 3.8.1).

a. Personal Air Lock (Breech Type)

The personnel air lock (Figure 3.8-20) consists of the air lock doors and the lock barrel. Its centerline is located at elevation 29'-6" and an azimuth of 315°, as shown on Figure 1.2-4.

Significant dimensions are as follows:

Parameter	Dimension
Clear Opening	7'-0"
O.D. of Flange on Door	7'-9 1/8"
Barrel Thickness	5/8"
Cover Thickness (Spherical Dished Head)	5/8"

The air lock barrel has a door on each end, each of which is designed to withstand the design pressure from inside the containment. The doors are hinged and swing away from the air lock barrel. Each door is fitted with two seals that are located so that the area between seals can be pressurized and tested to 52.0 psig. A leak chase system is provided over the barrel-liner joint of the personnel air lock for leak testing.

The locking device for the doors is a rotating, third ring, breach-type mechanism. These doors are interlocked so that only one door can be opened at a time. The capability exists for bypassing this interlock and relieving the internal pressure by use of special tools. The doors may be operated mechanically.

A sight glass is provided at the exterior of each door to permit observation of the status of the opposite door.

Separate limit switches are provided to allow remote indication of door position, and a signal is furnished for use by the control room.

The barrel, which is also the sleeve for the personnel air lock, is embedded and anchored in the shell of the concrete containment.

b. Equipment Hatch

The equipment hatch (Figure 3.8-21) consists of the barrel, the spherical dished cover plate with flange, and the air lock mounting sleeve. The centerline of the hatch is located at elevation 37' – 0 1/2" and an azimuth of 150°, as shown on Figure 1.2-4. The hatch opening has an inside diameter of 27' - 5".

A sleeve for a personnel air lock, the inside diameter of which is 9'-10", is provided at centerline elevation 30'-6". Thicknesses of the primary components are as follows:

Component	Thickness
Barrel	3 1/2"
Spherical Dished Cover Plate	1 3/8""
Flange	5 3/8"
Air Lock Mounting Sleeve	1 1/2"

The equipment hatch cover is fitted with two O-ring seals that enclose a space which can be pressurized and tested to 52.0 psig. The flange of the cover plate is attached to the hatch barrel with 32 swing bolts, 1 3/8" in diameter. A leak chase system is provided over the barrel-liner joint of the equipment hatch for leak testing.

The barrel, which is also the sleeve for the equipment hatch, is embedded in the shell of the concrete containment.

Provision has been made to lift the equipment hatch cover to the side to clear the opening and to store the cover in the storage saddles designed for this purpose.

A platform to allow access to the swing bolts at the top of the hatch has been permanently installed on the equipment hatch. Extensions have been added to the lower lifting lugs. These extension legs make the lifting device attachment points at approximately the same elevation for all four lug locations. The extension legs and the upper lifting lugs are used to support the hatch access platform. This platform is provided with fall protection chain railings and safety harness tie-off points.

Inserted into the mounting sleeve through the equipment hatch cover is a personnel air lock consisting of two air lock doors, two air lock bulkheads, and the air lock barrel.

Significant dimensions of the air lock are as follows:

Parameter	Dimension
Inside Diameter of Barrel	9'-6"
Barrel Thickness	1/2"
Door Opening	6'-8"x3'-6"
Door Thickness	3/4"
Bulkhead Thickness	1 1/8"

Each door is locked by a set of six latch pin assemblies, and is designed to withstand the design pressure from inside the containment. To resist the test pressure, each door is fitted with a set of test clamps. The doors are hinged and both swing into the containment. Each door is fitted with two seals that are located such that the area between seals can be pressurized and tested to 52.0 psig.

The doors are mechanically interlocked so that only one door can be opened at a time. The capability exists for bypassing this interlock and equalizing the pressure by use of special tools. The doors may be operated mechanically.

A sight glass is provided in each door to permit observation of the opposite door.

Separate limit switches are provided to allow remote indication of door position and to alarm in the control room.

c. Piping Penetrations

Piping penetrations are divided into two types, high energy and moderate energy. Moderate energy piping penetrations are used for process pipes in which both the pressure is less than or equal to 275 psi, and the temperature of the process fluid is less than or equal to 200°F. High energy piping penetrations are used for that piping in which the pressure or temperature exceeds these values.

High energy piping penetrations (Figure 3.8-22) consist of a section of process pipe with an integrally-forged flued head, a containment penetration sleeve and, where a pipe whip restraint is not provided, a penetration sliding support inside the containment. The sliding support provides shear restraint while permitting relative motion between the pipe and the support. The annular space between the process pipe and the sleeve is completely filled with fiberglass thermal insulation. The pipe and the flued head, are classified as ASME III, Safety Class 2 (NC), whereas the sleeve is classified as part of the concrete containment, ASME III (CC). The sliding support inside the containment is classified as an ASME Safety Class 2 component support (NF).

Moderate energy piping penetrations (Figure 3.8-23) consist of one or more process pipes, the containment penetration sleeve, and a flat circular end-plate. The pipe is classified as ASME III Safety Class 2 (NC). The sleeve is classified as ASME III Div. 2 (CC). The sleeve backed by concrete is ASME III (CC); the sleeve not backed by concrete is ASME III (Class 2). The end-plate material is classified as ASME Class 2.

Both kinds of assemblies, however, were analyzed to meet the intent of Class MC criteria (NE) through the use of a simplified, but conservative, approach described in Subsection 3.8.2.4.

d. Electrical Penetrations

Electrical penetrations (Figure 3.8-24) consist of a stainless steel header plate (bulkhead) with an attached terminal box, electrical modules which are clamped to the header plate, and a carbon steel weld ring which is welded to the header plate and to the sleeve. The metallic pressure resisting parts, the sleeve, stainless steel header plate and carbon steel weld ring were designed as ASME III Safety Class MC components (NE); that portion of the sleeve which is backed by concrete was designed as part of the concrete containment, ASME III (CC).

Double silicone and EPR O-rings provide a seal with a cavity for leakage monitoring between the header plate and the modules. The header plate is provided with a hole on the outside of the containment to allow for pressurization of the penetration assembly for leakage monitoring.

e. Instrumentation Penetrations

Instrumentation penetrations are of two types, electrical and fluid. The electrical type is similar in construction to the other electrical penetrations, and the discussion in Subsection 3.8.2.1d, as well as material appearing elsewhere in Subsection 3.8.2 which pertains to electrical penetrations, is applicable to these penetrations.

The second type of instrumentation penetrations, the fluid type, is similar in construction to the moderate energy piping penetrations. Consequently, the discussion in Subsection 3.8.2.1c on moderate energy piping penetration and other material in Subsection 3.8.2 relating to these penetrations are also applicable to this type of instrumentation penetration.

f. Fuel Transfer Tube Assembly

The fuel transfer tube assembly (Figure 3.8-25) consists of the fuel transfer tube, the penetration sleeve, the fixed saddle on the reactor side, and the sliding saddle in the Fuel Storage Building. The fuel transfer tube and its flanges were designed as ASME III Class 2 components (NC). The quick closure hatch on the refueling canal side of the tube was designed and fabricated to ASME Section III Class MC component (NE) requirements. That portion of the sleeve which is backed by concrete was designed as part of the concrete containment, ASME III (CC). The remaining pieces of the assembly were designed as ASME III Component Supports (NF).

g. Ventilation Penetration Assemblies

There are two types of ventilation penetrations, the containment air purge penetrations and the containment online purge penetrations. The containment air purge penetrations (Figure 3.8-26) each consist of a pipe sleeve (a rolled and welded pipe section, 36" O.D. by 1/2" wall thickness) which is flanged at each end with 36" weld neck flanges and, attached to these flanges, is a butterfly valve inside containment and a testable blind flange outside containment. Together with the pipe, the blind flanges form a part of the containment pressure boundary during plant Modes 1, 2, 3, and 4. During Modes 5 and 6, the blind flanges are replaced by spool pieces to configure the CAP system to perform its heating and ventilation functions. The valves are 36" diameter butterfly valves with fail-safe pneumatic operators. The weld between the pipe and the containment liner is equipped with a leak chase for pressure testing.

The containment online purge penetrations each consist of a pipe sleeve (a rolled and welded pipe section, 8" O.D. by 1/2" wall thickness). A short section of pipe with a nipple is welded to the sleeve on the outside of the containment, and a 3/4" valve and test connection is attached to it. The ends of this resulting assembly are welded to 8" weld neck flanges which are through-bolted to the inner and outer isolation valves. These valves are 8" diameter butterfly valves having fail-safe pneumatic operators. The weld between the pipe sleeve and the containment liner is equipped with a leak chase for pressure testing. Since the details, other than size, of these penetrations are essentially the same as those of the containment air purge penetrations, an additional figure is not included.

The portion of the penetration sleeve which is backed by concrete was designed as part of the concrete containment, ASME III (CC). That portion of the sleeve extending beyond the containment was designed to meet the requirements of ASME Section III, Division 1, Subsection NE, in addition to the fabrication and nondestructive examination requirements of Division 2. The valves and flanges were designed as ASME III Code Class 2 components (NC) and are further described in Subsection 3.9.3.

3.8.2.2 Applicable Codes, Standards, and Specifications

The design, materials, fabrication and inspection requirements for the major containment penetrations conform to, but are not necessarily limited to, the applicable sections of the following codes and specifications which are used to establish or implement design bases and methods, analytical techniques, material properties and quality control provisions.

Dates and revisions given for the listed codes are the earliest version that was used. Subsequent issues were incorporated into the design where practicable, or where the new issue directly affected the safety of the structure.

Code or Specification	Title
ASME Boiler & Pressure Vessel Code	Section II - Material Specification
	Section III, Division 1, Subsection NA, General Requirements
	Section III, Division 1, Subsection NB, Class 1 Components
	Section III, Division 1, Subsection NC, Class 2 Components
	Section III, Division 1, Subsection NE, Class MC Components
	Section III, Division 1, Subsection NF, Component Supports
	Section III, Division 2 Code for Concrete Reactor Vessels and Containments
	1975 Edition including Winter 1976 Addendum for Containment Liner
	Section IX - Welding Qualifications

Applicable Codes Dates for ASME B&PV Code, except Section III, Division 2, Components:

1971 Edition, through Summer 1973 Addenda (Equipment Hatch)

1974 Edition, through Winter 1975 Addenda (Personnel Air Lock in Equipment Hatch)

1974 Edition, no Addenda (Breech Type Personnel Air Lock)

1974 Edition, including Summer 1975 Addendum (High Energy Piping Penetrations)

1974 Edition, no Addenda (Moderate Energy Piping Penetrations)

1974 Edition, including Winter 1976 Addendum (Electrical Penetrations)

1971 Edition through Summer 1973 Addenda (Fuel Transfer Tube Assembly)

1971 Edition, including Winter 1973 Addendum (Ventilation Penetration Assemblies)

Code or Specification	Title
U.S. Department of Labor	Occupational Safety and Health Administration Standards, October 1975 Edition
UBC	International Conference of Building Officials, Uniform Building Code, 1973 Edition
AISC	Specification for the Design Fabrication and Erection of Structural Steel for Buildings, 1969 Edition, (including Supplements 1, 2 and 3)
ANSI B16.5-1968	Steel Pipe Flanges, Flanged Valves, and Fittings
ANSI N45.2-1974	Quality Assurance Program Requirements for Nuclear Power Plants
ANSI N45.4-1972	Leakage Rate Testing of Containment Structures for Nuclear Reactors
IEEE STD-317-1972	Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations

Code or Specification	Title
IEEE STD-323-1974	Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
IEEE STD-344-1971	Guide for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
SP-44	Manufacturer's Standardization Society Steel Pipe Line Flanges, 1975 Edition
10 CFR 50, App. J	Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors

NRC Regulatory Guides	Title
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Rev. 0, 6/73)
1.63	Electrical Penetration Assemblies in Containment Structures for Water Cooled Nuclear Power Plants (Rev. 2, 7/78)
1.84	Code Case Acceptability, ASME Section III Design and Fabrication (Rev. 15, 5/79)
1.85	Code Case Acceptability, ASME Section III Materials (Rev. 15, 5/79)
1.163	Performance-Based Containment Leak-Test Program (Rev. 0, 9/95)

The below-listed UE&C design and construction specifications applicable to the containment penetrations were prepared in accordance with applicable codes, quality control requirements and NRC Regulatory Guides.

UE&C Specifications	Title
9763.006-QAS-1	Quality Assurance Administrative and System Requirements
9763.006-QAS-3	Quality Assurance Administrative and System Requirements For Safety-Related Electrical Equipment
9763.006-MPS-1	Material and Processing Requirements for Nuclear Power Plant Components
9763.006-NSS-0185	Fuel Transfer Tube
9763.006-WS-1-NE	Requirements for Welding and Non-Destructive Examination for Nuclear Pressure Class MC Components
9763.006-SD-15-2	Seismic Requirements for Equipment Hatch and Personnel Air Lock
9763.006-SD-118-1	Seismic Requirements for Electrical Penetrations
9763.006-15-1	Containment Liner
9763.006-15-2	Containment Equipment Hatch and Personnel Locks

3.8.2.3 Loads and Load Combinations

The containment penetrations were designed to withstand all credible conditions of loading, including preoperational test loads, normal startup, operational and shutdown loads, severe environmental loads, extreme environmental loads, and abnormal loads. Design limits and load combinations are in accordance with Article NE-3000 of Division I and Regulatory Guide 1.57. The penetrations were evaluated for several combinations of loads to assure that the response of the components would remain within the limits prescribed in Subsection 3.8.2.5.

In the paragraphs that follow, these loads and load combinations are grouped according to the type of penetration.

a. Equipment Hatch and Personnel Air Lock

1. Design Loads

(a) Preoperational Test Loads

These are loads which are applied during the initial and any subsequent structural integrity or leak rate testing of the containment.

(1) Test Pressure (P_t)

The containment and components listed in Subsection 3.8.2.1 are pressurized to 115 percent of the design pressure to test their structural integrity, i.e., the test pressure is 60 psig with design pressure being 52 psig.

(2) Test Temperature (T_t)

The maximum and minimum temperatures inside the containment during the test are 100°F and 50°F, respectively. The temperature on the outside of the containment, considered to be the ambient temperature, is 0°F, the minimum temperature at the site. Any thermal loads occurring during test conditions are induced by the gradient between the containment and ambient temperatures.

(b) Normal Startup, Operational, and Shutdown Loads

Normal loads encountered during normal plant startup, operation and shutdown include the following:

(1) Dead Load (D)

This includes the weight of the component and appurtenances.

(2) Live Loads (L)

The live load on the raised floors of the personnel locks, representing personnel and equipment that may be moved through there, is taken as 100 pounds per square foot. There are no other live loads.

(3) Operational Thermal Loads (T_o)

The normal containment operating temperatures range from 50° to 120°F. Outside temperatures vary between 0°F and 90°F.

(4) Pressure Variation (P_v)

Differential pressure loads result from pressure variation either inside the containment or in the Containment Enclosure Building. This pressure variation is produced either by atmospheric fluctuations or by HVAC equipment. The internal pressure varies between -3.5 and +1.5 psig.

(c) Severe Environmental Loads

These are loads that would result from external conditions which could infrequently be encountered during the plant life. The following loads are considered in this category:

(1) Wind Load (W)

There is no wind load considered due to the presence of the containment enclosure.

(2) Operating Basis Earthquake (E_o)

These are the loads generated by the Operating Basis Earthquake, which is the earthquake that could reasonably be expected to occur at the plant site during the operating life of the plant. The applied accelerations, for static analysis, which are valid for rigid components (f > 33 Hz) are 0.580g for horizontal motion and 0.528g for vertical motion; damping is 2 percent for both accelerations. These accelerations represent the peak response of the containment cylinder at the elevation of the equipment hatch, the highest penetration in the containment. Only the actual dead load is considered in evaluating the seismic response forces.

(d) Extreme Environmental Loads

Extreme environmental loads are those loads which result from postulated events which are credible but highly improbable. The following loads are included in this category:

(1) Safe Shutdown Earthquake (E_s)

These are the loads generated by the Safe Shutdown Earthquake, which is the maximum potential earthquake that could occur in the vicinity of the site, based on geological and historical investigations. The applied accelerations, for static analysis, which are valid for rigid components (f > 33 Hz) are 0.861g for horizontal motion and 0.748g for vertical motion; damping is 3 percent for both accelerations. As with the operating basis earthquake, these accelerations also represent the peak response of the containment cylinder at the elevation of the equipment hatch.

Only the actual dead load is considered in evaluating the seismic response forces.

(2) Tornado Loads (W_t)

Due to the presence of the Containment Enclosure Building and other appurtenant structures, such as the pipe chases and the Fuel Storage Building, wind pressure and pressure variation are not considered in the design of containment penetrations.

Furthermore, due to various shield walls the equipment hatch and personnel air lock are protected from impact by tornado-generated missiles. All of the other penetrations (piping, ventilation, etc.), emerge from the containment inside the various appurtenant structures and, consequently, are also protected from impact by tornado-generated missiles.

(e) Abnormal Loads

Abnormal loads are those loads generated by postulated high energy pipe ruptures, particularly a rupture in the Reactor Cooling System resulting in a loss-of-coolant accident (LOCA).

(1) Accident Pressure (P_a)

The components were designed to withstand an internal containment accident pressure of 52.0 psig. See Subsection 6.2.1 for the development of this pressure.

(2) Accident Temperature (T_a)

The peak accident temperature of the liner plate is 268°F. However, a maximum temperature of 271°F was considered for the design of the components. See Figure 3.8-9, Figure 3.8-10 and Figure 3.8-11 for plots of the transient containment liner temperature response, as well as the transient containment pressure response.

(3) Internal Missile Loads

Internal missile loads, as described in Section 3.5, are prevented by shields, as required, which confine the missiles.

(f) Environmental Loads

These are environmental conditions which must be considered in regard to their effect on the durability of materials.

(1) Normal Relative Humidity

Under normal operating conditions, the relative humidity, both inside and outside the containment, is expected to vary between 5 percent and 100 percent.

(2) Accident Environmental Conditions

Following a loss-of-coolant accident, the following conditions will exist inside the containment.

a Relative humidity of 100 percent

b Spray solution of the following:

- Boron (calculated as boric acid), (minimum/maximum), 0.23/0.25 percent by weight
- pH (minimum/maximum), 9.0/9.6
- Sodium Hydroxide (minimum/maximum), 0.45/0.54 percent by weight

(3) Radiation

Time-integrated doses for personnel air locks and equipment hatch are as follows:

Personnel Air Locks and Equipment Hatch

Accident Doses	Inside Containment	Outside Containment
(1 year post accident)		
Gamma TID	$1.4x10^7$ rads	$6.5x10^4$ rads
Beta TID	$4.95x10^7$ rads	----
Normal Doses		
(40 years period)		
Gamma TID	$2x10^7$ rads	$5.3x10^3$ rads
Beta TID	Negligible	Negligible

(g) Fatigue Requirements

The personnel air locks and equipment hatch were designed to withstand the following conditions:

- 120 cycles of plant startup and shutdown
- 400 OBE cycles
- 100 SSE cycles
- 1 accident cycle (LOCA)
- 160 pressure test cycles

2. Load Combinations

The load combinations for the equipment hatch and personnel air locks are specified in Table 3.8-6.

b. High Energy Piping Penetrations

1. Loads

Design loads applied to high energy piping penetrations were based on selected percentages of plastic load capacities of the attached piping. Seismic loads were based on the actual seismic response of the containment and piping at the elevation of the penetrations. When actual piping loads were available, having been derived from analysis of the piping systems, they were compared with the design loads; whenever a final load exceeded its design value, a re-analysis was made to validate the final piping loads. Loads were divided into two categories, Design (which includes normal, upset and emergency conditions) and Faulted, based on operating conditions. Magnitudes of individual design loads are given below.

(a) Axial Load (N)

For Design Condition, axial load varied from 5 percent to 80 percent of pipe plastic axial load. For the Faulted Condition, 65 percent to 100 percent was used.

(b) Axial Load Due to SSE Conditions (N_s)

This load was based on the mass of the flued head, and the portion of the sleeve up to the containment wall (outside) and the seismic acceleration at the elevation of the penetration.

(c) Shear Load (V_1)

This is the shear applied to the flued head. For the Design condition shear load varied from 5 percent 80 percent of the pipe plastic shear load. For the Faulted condition, 60 percent to 100 percent was used.

(d) Shear Load (V_2)

This is the shear in the process pipe at the location of the sliding support. Load V_2 was taken as 10 percent of the pipe elastic shear for the design condition and 100 percent for the faulted condition. It was determined interactively with the bending moment (M).

(e) Seismic (SSE) Shear at Flued Head (V_s)

Load V_s is based on the mass of the flued head and the portion of the sleeve up to the outside of the containment wall and the seismic acceleration at the elevation of the penetration.

(f) Pressure Inside the Cavity (P_1)

The design pressure inside the cavity was taken as 56 psi, which conservatively represents the containment design pressure (52 psi).

(g) Pipe Pressure (P_2)

This is the internal design pressure in the process pipe.

(h) Torsion (T)

This was taken as 50 percent and 100 percent of the pipe plastic torsion load for the Design and Faulted loading conditions, respectively.

(i) Moment Loads (M_1 and M_2)

These moments were considered equal and were taken as percentages of the value of moment load calculated as per the ASME B&PV Code, Section III, Division 1, Subsection NC-3652.1 and the Standard Review Plan 3.6.2, Branch Technical Position MEB 3-1, Paragraph B(1) (b)(1)(e). For both the Design condition and the Faulted condition, the percentages taken ranged from 60 percent to 100 percent.

2. Load Combinations

All loads were considered simultaneously and were grouped into two categories. Design (which includes normal, upset and emergency conditions) and Faulted.

3. Environmental Conditions

(a) The normal temperature of the air within the containment enclosure outside the containment structure surrounding the penetration is 104°F maximum.

(b) The temperature of the air inside the containment structure is 120°F maximum (during normal and upset operating conditions).

(c) Loss of heat from the penetration to the air surrounding the penetration both outside and inside the containment structure is by natural convection.

(d) Process fluid temperatures were considered and vary from penetration to penetration.

(e) No effects of radiation were considered for high energy piping penetrations.

c. Moderate Energy Piping Penetrations

1. Loads

Loads applied to the penetration include actual piping loads, as determined from the piping analysis, and other applicable containment design loads. These loads are in compliance with the intent of Regulatory Guide 1.57 and are described below.

(a) Normal Axial Load (N)

This is the axial load imposed by the pipe onto the penetration.

(b) Shear Load (V)

This is the shear load imposed by the pipe onto the penetration.

(c) Torsional Load (T)

This is the torsional load imposed by the pipe onto the penetration.

(d) Bending Moment (M)

This is the bending moment imposed by the pipe onto the penetration.

(e) Pressure Inside the Cavity (P)

The design pressure inside the cavity was taken as 56 psig, which conservatively represents the containment design pressure (52 psig).

(f) Process Pipe Pressure Load (P_1)

This is the pressure inside the process pipe.

(g) Deadweight (D)

The deadweight of the penetrations and attachments is considered negligible.

(h) Temperature Load (T)

This is the load produced by thermal expansion of the attached piping.

(i) Operating Basis Earthquake Load (OBE)

These are the forces on the penetration from the attached piping caused by the operating basis earthquake acting on the pipe.

(j) Anchor Displacement Loads For the OBE (SAD OBE)

These are the forces produced by differential seismic displacements between the penetration and the piping anchors due to the OBE.

(k) Thrust Load (TH)

This is the load applied along the axis of the penetration.

(l) Safe Shutdown Earthquake Load (SSE)

These are the forces on the penetration from the attached piping caused by the safe shutdown earthquake acting on the pipe.

(m) Anchor Displacement Loads for the SSE (SAD SSE)

These are the forces produced from the attached piping by differential seismic displacements between the penetration and the piping anchors due to the SSE.

(n) Pressure and Thermal Anchor Displacement Load (PAD + TAD)

These are the forces on the penetration caused by differential displacement between the penetration and the piping anchors due to pressure and thermal loads.

(o) Pipe Rupture Load (PR)

This is the load on the penetration caused by the rupture of an attached pipe.

2. Load Combinations

Three design loading conditions were investigated: the Design/Normal Condition (Service Level A), the Upset Condition (Service Level B), and the Emergency/Faulted Condition (Service Level C/D). Load combinations used for each of these design loading conditions are presented in Table 3.8-7.

d. Electrical Penetrations

1. Loads

(a) Dead Load (D)

The weight of the penetration and its attachments were considered.

(b) Test Pressure (P_t)

The maximum test pressure inside the containment was 60.0 psig. This is the largest pressure to which the penetrations are subjected and, to simplify the calculations, this is the only pressure that is used for analysis.

(c) Accident Pressure (P_a)

The maximum containment pressure occurring during a design basis event is 52 psig.

(d) Test Temperature (T_t)

The temperature inside the containment during the pressure test will vary between 50°F and 100°F.

(e) Operational Temperature (T_o)

Under normal operation the temperature inside the containment will vary between 50°F and 120°F.

(f) Accident Temperature (T_a)

A peak accident temperature of 271°F was utilized for the design of components. This is a design temperature which is higher than the actual temperature to which the penetrations are subjected, and, to simplify the calculation, all material properties are based on this temperature. See Figure 3.8-27 for a plot of the actual transient temperature response of the electrical penetration seal.

(g) Operating Basis Earthquake (E)

A static equivalent load factor of 4.0 was used for both horizontal and vertical earthquakes. This value exceeded all acceleration values calculated in the response spectra at the elevation of the penetrations.

(h) Safe Shutdown Earthquake (E_{ss})

For the SSE, a static equivalent load factor of 4.0 was also used for both horizontal and vertical earthquakes. This value exceeded all acceleration values calculated in the response spectra at the elevation of the penetrations.

(i) External Pressure (P_e)

Under normal operation, the containment internal pressure varies between +1.5 psig and -3.5 psig. Thus, the maximum "external" pressure during normal operation was 3.5 psig.

2. Environmental Conditions

(a) Outside Temperature

The electrical penetrations were designed for temperatures outside the containment which vary between 40°F and 104°F. Following a DBA, the maximum temperature is 130°F.

(b) Atmospheric Pressure

Pressure on the outside of the containment varies between +0.006 psig and -0.005 psig during normal operation and between 0 psig and -0.011 psig during the containment test. Following a DBE, the minimum pressure is -0.013 psig.

(c) Relative Humidity

The maximum relative humidity inside the containment is 90 percent and outside the containment is 95 percent. Following a DBE, the relative humidity inside the containment is 100 percent.

(d) Radiation

The normal radiation rate is 50 millirads per hour. During the containment test, the radiation rate is 15 millirads per hour. The total cumulative radiation, considering 40 years of normal operation, a DBE and the post DBE period, is $1.3\text{x}10^8$ Rads.

(e) Chemical Sprays

Chemical sprays include the following:

- Boric Acid	0.21 (by weight)
- pH (min/max)	8.5/10.5
- Sodium Hydroxide Solution	0.42 (by weight)

3. Loading Combinations

Loading combinations are given in Table 3.8-8.

e. Fuel Transfer Tube Assembly

1. Design Loads

(a) Preoperational Test Loads

(1) Test Pressure (P_t)

The test pressure is 60 psig.

(2) Test Temperature (T_t)

The temperature range inside the containment is 50°F to 100°F, and the minimum temperature at the site is 0°F.

(b) Normal Loads

(1) Dead Load (D)

This includes the weight of the fuel transfer tube and its support system.

(2) Live Load (L)

This includes the weight of the fuel assembly and carriage and the weight of water in the fuel transfer tube.

(3) Pressure Variation (P_v)

The containment internal pressure varies between -3.5 and +1.5 psig.

(4) Hydrostatic Pressure (H)

The maximum depth of water, above the tube, during refueling is 34'-4¼".

(c) Severe Environmental Loads

(1) Wind Loads (W)

There is no wind load due to the presence of the Fuel Storage Building.

(2) Operating Basis Earthquake (E_o)

The accelerations produced by the operating basis earthquake, which are valid for rigid components (f >33 Hertz), are 0.45g for N-S motion, 0.79g for E-W motion, and 0.42g for vertical motion.

The OBE seismic loads also included the forces produced by differential seismic displacement between the Containment Building and the Fuel Storage Building.

(d) Extreme Environmental Loads

(1) Tornado Loads (W_t)

Tornado loads are not considered due to the presence of the Fuel Storage Building.

(2) Safe Shutdown Earthquake (E_s)

The accelerations produced by the safe shutdown earthquake, which are valid for rigid components (f >33 Hertz), are 0.75g for N-S motion, 1.12g for E-W motion, and 0.61g for vertical motion.

The SSE seismic loads also included the forces produced by differential seismic displacement between the Containment Building and the Fuel Storage Building.

(e) Abnormal Loads

(1) Accident Pressure (P_a)

The containment internal accident pressure of 52.0 psig was considered in the design of the fuel transfer tube assembly.

(2) Accident Temperature (T_a)

A peak accident temperature of 271°F was considered for the design of the components.

(f) Fatigue Requirements

The fuel transfer tube was designed to withstand the following conditions:

- 400 OBE cycles
- 1 accident cycle (LOCA)
- 160 pressure test cycles
- 1000 temperature cycles

2. Load Combinations

The load combinations are given in Table 3.8-9.

f. Ventilation Penetration Assemblies

1. Design Loads

(a) Deadweight (D)

The deadweights of the sleeve, valves and attached piping, as applicable, were included.

(b) Operating Basis Earthquake (OBE)

The inertia forces produced by the operating basis earthquake were included.

(c) Safe Shutdown Earthquake (SSE)

The inertia forces produced by the safe shutdown earthquake were included.

(d) Normal Containment Pressure (P_n)

The normal containment pressure varies between -3.5 and +1.5 psig.

(e) Accident Pressure (P_f)

The accident pressure on the containment due to either a LOCA or a steam line break, whichever is higher, is used. The LOCA pressure governs and is 52.0 psig for design purposes.

(f) Test Pressure (P_t)

The containment test pressure is 60.0 psig.

2. Load Combinations

(a) Normal and Upset

$D + OBE + P_n$

(b) Faulted

$D + SSE + P_f$

(c) Test

$D + P_t$

3.8.2.4 Design and Analysis Procedures

The portions of the containment classified as steel containment were designed and analyzed using procedures described below. The components were designed to safely withstand the load combinations defined in Subsection 3.8.2.3.

a. Personnel Air Lock

The personnel air lock was designed and analyzed in accordance with Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1.

The air lock door was designed for an internal pressure of 52 psi combined with an equivalent pressure due to the OBE (0.40 psi) and the SSE (0.60 psi). The total pressure was applied to the complete cover. Design was governed by the SSE load in the Faulted Condition; hand computations were used.

That portion of the air lock barrel not backed by concrete was analyzed by hand computations for the most critical load combination in the Faulted Condition. The internal pressure and an additional pressure representing the effect of horizontal seismic loading result in a total pressure of 52.54 psi on the vertical projection of the shell. The deadweight and vertical seismic loading result in a vertical load of 20,277 lbs.

Stresses and strains within the barrel due to displacements imposed by the concrete were determined using a finite element model and the computer program ANSYS described in Subsection 3.8.2.4i. The displacements of the concrete were determined in the three-dimensional finite element analyses described in Subsection 3.8.1.4. Both elastic and elastic-plastic behaviors were considered.

Using hand computations, it was also determined that the natural frequencies associated with axial vibration of the air lock door and of the barrel as well as transverse vibration of the barrel are all much greater than 33 Hertz. Therefore, the personnel air lock was considered rigid for purposes of seismic analysis. The natural frequency for vertical vibration of the door in the open position, however, is less than 33 Hertz. This condition is not under the jurisdiction of Subsection NE but was considered for the design of the hinge plates and pins.

b. Equipment Hatch

The equipment hatch was designed and analyzed in accordance with Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1.

The equipment hatch cover was designed for the effects of accident pressure combined with the SSE, with the total applied pressure being 52.40 psi. Since live load and dead load are negligible, this combination represents the Faulted Condition with SSE, load combination 5 in Table 3.8-6.

The flange, bolts, and other components were also designed for the above indicated load combination as well as for external pressure, bolt preload, and other loads produced by gasket seating and seal testing. Hand computations were used.

Rotation of the flange was calculated to verify seal integrity, using an axisymmetric finite element model and the computer program AX2 (described in Subsection 3.8.2.4i). The critical loads for this analysis are bolt preload, with pressure on the convex side of the head, and bolt preload acting alone. In both cases, the integrity of the seal was not violated.

The design of that portion of the barrel not backed by concrete was based on analysis for accident pressure and for test pressure. Dead load and seismic load are negligible and were omitted. An axisymmetric shell model was analyzed using the computer program ANSYS. In these analyses, two sets of boundary conditions were used, one in which the flange was considered rigidly attached to the barrel and another in which the far end of the barrel was free. In this manner, the actual restraint was bounded.

A similar finite element model was used to determine the minimum thickness required for the end portion of the barrel, which is backed by concrete. Results at the concrete-to-metal junction verified the adequacy of a thickness of 1 inch.

Buckling of the head, sleeve, and barrel was checked, using the procedures of Subarticle NE-3133 of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, and found not to occur. Fatigue in the barrel was also investigated, and the design was found to be satisfactory.

To verify the design of the entire equipment hatch assembly, displacements imposed by the concrete onto the barrel were considered in an axisymmetric shell model which can accept nonaxisymmetric loading. The entire assembly, cover, flange and barrel, were modeled using ANSYS. Displacements were imposed at the intersection of the barrel with the containment liner, and accident pressure and temperature, dead load, live load, and seismic load were also included. Note that the personnel air lock was omitted from this model because simplified analyses of that region showed it to be similar in stiffness to the equipment hatch. There were four sets of imposed displacements based on four different cracking patterns in the concrete. For all load combinations, strains were added on an elastic basis and found to be below allowable values.

The air lock mounting sleeve, including its reinforcement, was analyzed by hand computation. Loads considered included accident pressure, dead load, live load, seismic load, and steady-state temperature. Stress analysis of the sleeve at the sleeve-collar boundary was performed using the AX2 axisymmetric shell program, considering only test pressure.

Transverse natural frequencies of the hatch, sleeve, and air lock were all found by hand computations to be greater than 33 Hertz. Using axisymmetric shell elements and the computer program ANSYS, the axisymmetric modes of vibration of the hatch were also found to be greater than 33 Hertz. Therefore, the equipment hatch assembly was considered rigid for purposes of seismic analysis.

c. Personnel Air Lock in Equipment Hatch

The personnel air lock in the equipment hatch was designed and analyzed in accordance with Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1.

This air lock was analyzed for accident pressure and SSE seismic loads. Overall structural response was calculated using the computer program STARDYNE (described in Subsection 3.8.1.4) and a three-dimensional model. Localized stresses at attachments, such as latches, pins and bearings, were determined using hand computations. In the analysis of the latches, an external pressure of (-)3.5 psig was applied in conjunction with the seismic loads.

The personnel air lock floor was designed for live load and dead load, and the test clamps, bolts, and plates were designed for the test pressure, all using hand computations. The open door seismic condition was also checked, considering the door in this position to be seismically flexible.

However, for all other design conditions the natural frequencies of the air lock assembly were found to be greater that 33 Hertz, and the assembly was considered rigid for purposes of seismic analysis. These calculations for natural frequency were made using the same finite element model that was used to determine overall response of the air lock.

d. High Energy Piping Penetrations

High energy piping penetrations were designed and analyzed to meet the intent of Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1. The design insured that the sleeve have a greater load carrying capacity than the process pipes.

The analysis procedure utilized three phases.

The first phase consisted of a heat transfer analysis to determine the temperature in the flued head, the sleeve and at the sleeve/containment wall interface. This was accomplished through the use of a finite element model and the computer program MARC-CDC (described in Subsection 3.8.1.4).

After establishing that concrete temperatures complied with the limit of ASME Section II, Division 2, Subsection CC-3340(b), a finite element model for stress analyses was established. This second phase consisted of a stress analysis utilizing unit load cases representing axial (N), shear (V), torsion (T), and moment (M) loadings.

Two additional load cases were run, using the containment design pressure and the actual thermal gradient, as determined in the heat transfer analysis. Total stresses for the design loads presented in Subsection 3.8.2.3 were determined by multiplying stresses from the unit load cases by the appropriate load ratio and combining the results with those from the pressure and thermal gradient runs.

The third phase consisted of using stress results from phase two to determine stress intensities at various locations in the sleeve and flued head and comparing these intensities to allowable values.

Computer programs used for stress analysis included MARC-CDC, WILSON 1 and WILSON 2 (all described in Subsection 3.8.1.4), and ANSYS (described in Subsection 3.8.2.4i). The temperature gradient which was used in the stress analyses was determined using the computer program TAPAS (described in Subsection 3.8.1.4); thermal boundary conditions were determined in the MARC-CDC heat transfer analysis of phase one.

e. Moderate Energy Piping Penetrations

Moderate energy piping penetrations were designed and analyzed to meet the intent of Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1. The design insured that the sleeves have a greater load carrying capacity than the process pipes.

The analysis procedure consisted of subjecting the penetration assembly to actual pipe loads and other applicable containment loadings and determining stresses at four critical locations, which are the junctions of the sleeve with the containment wall and the end plate and the junctions of the end plate with the sleeve and the process pipe. Weld stresses at the pipe/end plate interface weld were also determined.

All stresses were compared with appropriate allowable values to determine the adequacy of the penetrations.

All computations were performed by hand using standard formulas and techniques taken from established references and/or by the finite element program ANSYS.

f. Electrical Penetrations

Electrical penetrations were designed and analyzed in accordance with Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1.

A typical penetration was analyzed for dead load, test pressure (being the larger of maximum LOCA pressure and test pressure) and safe shutdown earthquake loads. The temperature was taken as 375°F for all loading conditions.

It was considered unnecessary to do a thermal analysis since the maximum temperature the assembly will be exposed to is only 370°F. Fatigue analysis was also unnecessary since it is expected that only a few cycles of loading will be experienced in the 40-year life of the plant.

Using appropriate hand computations, the below-listed penetration components were analyzed. Items 1 through 9 comprise the pressure boundary and Items 1 through 8 code (Division 1) items. Items 10 and 11 are accessories which were also analyzed.

Item 1 - Installation weld which attaches the penetration assembly to the nozzle

Item 2 - 1/2-13 stud which holds the single clamp in place

Item 3 - 5/8-11 stud which holds the center clamp in place

Item 4 - Module - The module transmits the pressure applied over its projected area to the clamps.

Item 5 - Monitoring plate

Item 6 - Attaching weld (Monitor plate to flange extension)

Item 7 - Center clamp which receives loading from three modules. Each module contributed 1/3 of the load applied to it.

Item 8 - Single clamp. Each module contributed 1/3 the load applied to it.

Item 9 - Epoxy sealant material in the penetration module

Item 10 - Penetration terminal box (outboard)

Item 11 - Penetration terminal box (inboard).

To further verify the integrity of the penetration, it is pressure tested as described in Subsection 3.8.2.7.

g. Fuel Transfer Tube Assembly

The fuel transfer tube and its support system were designed and analyzed in accordance with Subsection NC of the ASME Boiler and Pressure Vessel Code, Section III, Division 1.

The fuel transfer tube assembly was analyzed for the various load combinations as listed in Table 3.8-9. Overall structural response was calculated using the computer program STARDYNE and a three-dimensional finite element model. This model was also used to determine the natural frequencies of the assembly, the lowest of which was 34.14 Hertz. Therefore, the fuel transfer tube assembly was considered rigid for purposes of seismic analysis.

Buckling of the tube was also checked, using the procedures of Subarticle NC-3133 of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC, and found not to occur.

In the load combinations which include seismic loads, the relative displacements due to differential movements between the Containment Building and the Fuel Storage Building were imposed at the sliding support.

h. Ventilation Penetration Assemblies

Ventilation penetrations were designed and analyzed to meet the intent of Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1. The design was based on system requirements, and subsequent analysis indicated that stress levels were well below an acceptable limit.

The analysis consisted of conservative static analysis procedures (hand computations) to determine stress levels at the interface with the outside of the containment wall, the critical region of the assemblies. Both primary and secondary stresses were calculated.

i. Descriptions of Computer Programs Utilized in the Design and Analysis

The computer programs used in the design and analysis of the steel components (Section III, Division 1 items) of the concrete containment which resist pressure and are not backed by concrete, are described briefly in this subsection. Programs listed below are only those programs which have not been previously described in Subsection 3.8.1.4g. A summary of the comparisons of results used to validate them is given in Appendix 3F. The program classifications are discussed in Subsection 3.8.1.4g.

1. ANSYS: Engineering Analysis System, by Swanson Analysis System, Inc. (Houston, Pa.). Documentation is available from Control Data Corporation. ANSYS provides static and dynamic analysis capabilities, including plasticity, creep and swelling; small and large deflections; steady-state and transient heat transfer; and steady fluid flow.

2. AX2: Axisymmetric Shell Program, by Pittsburgh-Des Moines (PDM) Steel Company. Documentation is available from PDM. AX2 is a computer program for the analysis of single layer, axisymmetric thin shells of revolution. Materials are isotropic and elastic, and loads are static and axisymmetric.

3.8.2.5 Structural Acceptance Criteria

The components classified as steel containment were designed to remain within the design limits specified in Subsection NE of the ASME B&PV Code, Section III, Division 1, and NRC Regulatory Guide 1.57.

Code boundaries for all penetrations except the personnel air lock are the portion of the sleeve which is backed by concrete and the attachment weld to the liner are Division 2; all remaining portions of the penetrations are Division 1. The division boundaries for the personnel air lock are described in Subsection 3.8.2.6c.

Additional acceptance criteria pertaining to specific penetrations are described below.

a. Equipment Hatch and Personnel Air Locks

Allowable stresses for the equipment hatch and personnel air locks are given in Table 3.8-10 and Table 3.8-11.

b. High Energy Piping Penetrations

The high energy piping penetrations were classified as ASME Section III, Division 1, Safety Class 2 components (NC). However, since the stress limits of Subsection NC are not directly applicable to this type of component for the specified categories of operating conditions, the penetrations were analyzed to meet the intent of the requirements of ASME Section III, Division 1, Subsection NB, and the allowable stress intensities were limited to the values given in Subsection NC. These allowables are summarized, in general terms, in Table 3.8-12.

c. Moderate Energy Piping Penetrations

The moderate energy piping penetrations were designed to meet the intent of ASME Section III, Division 1, Class MC (NE) criteria for stress limits and load combinations. Stress limits, used to determine the acceptability of the penetrations, are as follows:

	Primary Stress Intensity		Primary & Secondary Stress Intensity
	Membrane	Membrane + Bending	
Design/Normal Conditions	1.0 S_{mc}	1.5 S_{mc}	3 S_{mi}
Upset Conditions	1.0 S_{mc}	1.5 S_{mi}	3 S_{mi}
Emergency/Faulted Conditions	1.2 S_{mc} 1.0 S_y	1.8 S_{mc} 1.5 S_y	NA

d. Electrical Penetrations

All stresses are classified in accordance with Table NE-3217-1 of the ASME Code, Section III, Division 1, Subsection NE, and maximum stresses were found to be within the following limits:

1. The design limit of Subsection NE-6322 was not exceeded when the assembly was subjected to test pressure.

2. The design limits of Subsection NE-3131 (a), (b), and (d) were not exceeded when the assembly was subjected to the effects of the load combinations for test, normal, upset and faulted (abnormal/severe environmental) conditions.

3. The design limits of Subsection NE-3131 (c) (1) or (2), as applicable, were not exceeded when the assembly was subjected to the effects of the faulted (abnormal/extreme environmental) load condition.

4. The design limits of Subsection NE-3131.1 were not exceeded when the assembly was subjected to concurrently applied design loadings that produced the greatest potential for shell instability and loadings associated with the vibratory motion of the safe shutdown earthquake.

e. Fuel Transfer Tube Assembly

The fuel transfer tube assembly was designed to meet the intent of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC, for Class 2 components. The maximum stresses and displacements at the bellows were found to be within the following limits:

1. The design limits of Subsection NC-3200, Appendix XIII-1140 were not exceeded when the assembly was subjected to the effects of the load combinations representing test, normal, refueling and upset conditions.

2. The design limits of Subsection NC-3200, Appendix XIII-1140 were not exceeded when the assembly was subjected to the effects of the faulted load condition.

3. The fuel transfer tube was analyzed in accordance with Subsection NC-3133 for the cylindrical shell subject to test pressure, accident pressure and seismic loadings produced by the SSE.

4. The deflection capabilities of the bellows were not exceeded when the assembly was subjected to the effects of the load combinations for test, normal, refueling, upset and faulted conditions.

f. Ventilation Penetration Assemblies

The ventilation penetrations were designed to meet the intent of ASME Section III, Division 1, Class MC (NE) criteria for stress limits and load combinations. For all of the indicated load conditions the stress limit was 1.0 S_m.

3.8.2.6 Materials, Quality Control and Special Construction Techniques

a. Materials

Materials used for components classified as steel containment include steel plate, bolts, pins, seamless pipe, seals, welding filler materials, coatings, and other miscellaneous items. These materials are listed in Table 3.8-11 for the equipment hatch and personnel air locks, and in Table 3.8-13 for the piping, electrical, and instrumentation penetrations, fuel transfer tube and ventilation penetration assemblies. Not listed in these tables is the protective coating system which is identical to the system described in Subsection 3.8.1.6. These coatings are applied to all exposed surfaces.

Replaceable silicone seals are used in the equipment hatch and personnel air lock.

b. Special Construction Techniques

The only special construction technique employed was the manner in which the personnel air lock was fabricated. To maintain the construction schedule, the barrel, inside bulkhead and outside bulkhead were shipped to the site and assembled and tested in place. The testing described in Subsection 3.8.2.7 verified that this assembly procedure did not affect the integrity of the lock.

The equipment hatch was also fabricated in one piece to minimize field welding. Shop and field testing for the hatch are described in Subsection 3.8.2.7.

c. Quality Control

1. Equipment Hatch and Personnel Air Locks

The contractor certified by application of the appropriate code symbol and completion of the appropriate Data Report in accordance with Article NE-8000 (Division 1) that the material used complies with the requirements of Article NE-2000 (Division 1), and that the fabrication, installation, and construction comply with the requirements of Article NE-4000 (Division 1). Specific quality control requirements are further discussed below.

Quality control procedures were in accordance with Articles NE-4000 and NE-5000 of Subsection NE of the ASME Code, Section III, Division 1. For all components, including welding and brazing materials covered under this section, the fabricators supplied Certified Materials Test Reports. These reports included chemical analyses, physical tests, mechanical tests, examinations and heat treatment.

Physical tests include a Charpy-V-Notch toughness test performed in accordance with Article NE-2300 of Division 1. The test temperature was (-)10°F in the heat-affected zone and (-)25°F in the parent metal for the equipment hatch and personnel air locks.

Longitudinal seams in welded pipe were 100 percent radiographed.

Dimensional standards for individual metallic components were in accordance with Section NE-2700 (Division 1).

All welding was performed in accordance with NE-4000 (Division 1). When the base metal temperature was less than 50°F, welding was not performed without heating the metal. When base metal temperature was within the range of 0°F to 50°F inclusive, the base metal within 3" of the place where welding was to be started was heated to a minimum temperature of 50°F.

Fabrication of the equipment hatch was performed in accordance with NE-4000 (Division 1) using the best shop practices relative to edge treatment, alignment, and general workmanship.

Due to its field fabrication, the personnel air lock was stamped as follows: the center barrel from 8" inside the containment to 1'-3" outside the containment is NPT Class CC, Division 2, 1975, with the Data Report noting that parts extending beyond the concrete were fabricated in accordance with Division 1, 1974 Edition. The other two end sections were stamped NPT Class 2 or MC Division 1, 1974 Edition.

Fabrication tolerances of linear dimensions, such as length and diameter of hatch and locks, were as listed below unless specified otherwise on drawings.

Dimensions less that 6 inches	±1/32 inch
Dimensions 6 inches to 2 feet-6 inches	±1/16 inch
Dimensions over 2 feet-6 inches to 10	±1/8 inch
Dimensions greater than 10 feet	±1/4 inch

Erection tolerances insured that the equipment hatch and personnel air locks were set plumb, square and level and at their proper elevation and plane. Angular tolerance through the containment wall, as measured from the point of attachment of each air lock or the equipment hatch to the containment liner, was not greater than ±30 minutes. Angular tolerance of the theoretical point of attachment to the containment liner, as measured from the center of the containment, was ±4 minutes in the horizontal plane. Vertical tolerance was ±1 inch. Negative radial deviation from the theoretical vertical centerline did not exceed (-)¼" at the equipment hatch.

2. Other Penetrations

   Fabrication criteria for the other penetrations have been discussed in Subsection 3.8.2.1 with the descriptions of the penetrations.

3.8.2.7 Testing and In-Service Surveillance Requirements

a. Equipment Hatch and Personnel Air Locks

The structural testing and in-service inspection program consists of the following:

1. Strength test of personnel locks
2. Leak test of personnel locks
3. Leak test of leak chase system
4. Leak test of seals
5. Joint inspection
6. Other testing as described in Subsection 3.8.1.7.

Procedures are described in the paragraphs which follow.

The strength test is conducted in accordance with NE-6320 (Division 1); it consists of pressurizing the lock to 60.0 psig and then holding the pressure for 15 minutes with hold-down bars on the inner door.

Following the strength test, a leak test is conducted in accordance with NE-6215 (Division 1). During this test, the air pressure is reduced to 52.0 psig and held for 15 minutes. If a pressure drop occurs, soap bubble tests are conducted, repairs made, and the personnel lock retested.

Leak chase systems are leak tested in accordance with NE-6215 (Division 1) by pressurizing to 52.0 psig and holding for 15 minutes. If a pressure drop occurs, soap bubble tests are conducted, repairs made, and the personnel lock retested.

Seals for personnel lock doors, equipment hatch and at other locations are leak tested by pressurizing the space between the seals to 52 psig and holding for 15 minutes. If a pressure drop occurs, a halogen diode detector test was conducted, repairs made, and the seals retested.

All joints not covered by test channels are accessible for in-service inspection.

Testing of the entire containment structure, which includes the equipment hatch and personnel air lock, includes the integrated leak rate test, the in-service leak rate test, the preoperational structural integrity test, and general visual inspection of structurally critical areas. All of these tests are described in either Subsection 3.8.1.7 or in Subsection 6.2.6.

b. Other Penetrations

The portions of all the other penetrations which form part of the containment pressure boundary are subjected to the test program performed on the completed containment. This program includes the integrated leak rate test, the in-service leak rate test, the pre-operational structural integrity test and general visual inspection of structurally critical areas, all of which are described in either Subsection 3.8.1.7 or Subsection 6.2.6.

The penetrations also receive certain design verification testing as described below.

A typical electrical penetration is pressure tested in order to demonstrate that the module, epoxy and "O" ring seal can withstand the required pressure. A pressure chamber is fitted with the module and the pressure is raised to 6,000 psig without perceptible damage to the "O" rings or the penetration module. Thus a safety factor of 100 can be assumed (since the maximum design pressure is 60 psig).

3.8.3 Concrete and Structural Steel Internal Structures

The following section contains the physical description, codes, loads and load combinations, design and analysis procedures, allowable stresses, quality control and testing as they relate to the internal structures of the containment.

3.8.3.1 Description of the Internal Structures

The description of the structural configuration, materials, location and arrangement of the internal structures appears on the general arrangement drawings, Figure 1.2-2, Figure 1.2-3, Figure 1.2-4, Figure 1.2-5, and Figure 1.2-6.

The following major internal structures are located in the containment structure. They incorporate no unique or new design or construction features. Also, block or concrete masonry partitions are not utilized in the containment.

a. Reactor Support System

The reactor vessel is supported by two distinct support systems, the vertical box support and the lateral ring girder support. The vertical box support transmits vertical loads from the reactor vessel to the primary shield wall; it also transmits lateral loads from the reactor vessel to the lateral ring girder, which then transmits the loads to the primary shield wall. Design and analysis of the vertical box support are described in Subsection 5.4.14. The lateral ring girder support is described below.

The ring girder, reactor vessel lateral support, consists of a set of four curved girders resting on the primary shield wall at Elevation (-)14'-1", constructed of welded plates of ASTM A588, Grade 50 steel. As indicated above, its function is to restrain the lateral movement of the reactor vessel and to transfer the resultant lateral loads to the primary shield wall through normal contact. For details see Figure 3.8-28.

b. Steam Generator Support System

The steam generators are supported by several steel component supports, including vertical supports, upper lateral supports, and lower lateral supports. The vertical supports transmit loads to the fill mat and the upper and lower lateral supports transmit loads to both the primary and secondary shield walls. See Subsection 5.4.14 for a description of these supports.

c. Reactor Coolant Pump Supports

Reactor coolant pump supports, including vertical columns and lateral tension tie bars, are described in Subsection 5.4.14.

d. Primary Shield Wall

The primary shield wall is a circular reinforced concrete wall, varying in thickness from 4'-9" to 8'-6", enclosing and supporting the reactor pressure vessel. It is supported on the fill mat slab and extends to the refueling canal bottom elevation. In addition to providing shielding to the interior containment during normal operation or maintenance, the wall protects the reactor vessel from blowdown effects in the event of a rupture in the primary system piping outside the wall.

The reinforcing consists of several layers of hoop and meridional bars with the meridional bars extending either into the fill mat or the continuation of the fill mat down in the cavity in order to provide anchorage for the wall. See Figure 3.8-29.

e. Secondary Shield Wall

The secondary shield wall is a 4-foot thick octagon shaped reinforced concrete wall enclosing the reactor coolant piping, steam generators, reactor coolant pumps and their supports. The shield wall has openings to vent the area and to permit access from the outer annular area of the containment structure. These openings are protected by offset reinforced concrete walls which provide radiation shielding and protection from pipe whip. The secondary shield wall is supported on the fill mat slab and the above described portion extends to the underside of the operating floor.

Extending upward from the operating floor are those portions of the secondary shield wall that are referred to as the biological shield walls. They extend 7 feet above the operating floor and are shaped, in plan, like an elongated octahedron. Each one encloses two steam generators and provides radiation shielding.

f. Refueling Canal

The refueling canal consists of reinforced concrete walls and floors lined with ASTM A240 Type 304 stainless steel to provide a leak-tight membrane during refueling operations. The floor extends from the reactor cavity to the fuel transfer penetration and is supported on the primary shield wall and support walls which extend to the fill mat. The walls extend from the base of the canal to the operating floor level. During refueling the new and spent fuel elements are transported through the canal, which is flooded to Elevation 23.5 feet. Tools used during refueling are supported from the canal walls. For a more detailed description of the refueling canal and its use during refueling, see Subsection 9.1.4.

g. Pressurizer Region

The pressurizer region consists of a shield wall which provides shielding and missile protection. The wall is composed of several precast concrete sections which are placed one atop another. To maintain continuity these sections are bolted together. These sections are provided with lifting lugs and, when unbolted, may be removed to permit access to the pressurizer for maintenance and in-service inspection.

Pressurizer supports are described in Subsection 5.4.14.

h. Control Rod Drive Mechanism (CRDM) Missile Shield and Cable Bridges

The CRDM missile shield has been incorporated into the simplified head assembly. The original missile shield structure has been removed from the Containment. To support the CRDM and Drive Rod Position Indication (DRPI) System cables that run from the simplified head assembly two hinged cable bridges are provided on the operating deck, one on each side of the refueling cavity. See Figure 1.2-6A.

i. Operating Floor Slab

The operating floor slab is a reinforced concrete slab, 3'-0" to 8'-6" thick, located at Elevation +25.0 feet spanning between the crane support columns and the secondary shield walls. There are openings in the floor for access, installation, and removal of the four steam generators and the pressurizer, and access to the refueling canal and internal storage area. Removable reinforced concrete plugs are provided over the reactor coolant pumps and the in-core detector drive. The polar crane rail is located at the extremities of the operating floor concrete slab over its support structure.

j. Fill Mat

The internal structures are supported on a 4-foot thick reinforced concrete fill mat poured over the liner on the structural base mat. This fill mat is not connected to the base mat. Horizontal load transfer is provided by the keying action of the fill mat in the reactor pit, elevator pit, and sump pits. The fill mat extends under the secondary shield wall and the crane support columns to the containment cylindrical wall. An expansion joint is provided between the end of the fill mat and the containment cylindrical wall. The fill mat is thus independent of the primary containment structure walls. All supports for primary equipment and for other components and equipment located in the annulus area outside the secondary shield wall are supported on the fill mat.

k. Structural Steel

Structural steel is provided to support floors in the annulus at the operating level and in other areas of the containment structure where access to components and equipment is required, to provide a means of travel from various quadrants of the containment, and for pipe support at lower levels. Steel grating and concrete slabs are provided for a walking surface, and stairways and an elevator are required for movement between various levels of the annular steel.

l. Crane Support Structure

The crane rail is located on a continuous reinforced concrete beam at the operating floor level. The beam is supported on a series of reinforced concrete columns which extend to the fill mat outside the secondary shield wall. The beam is also connected to the biological shield wall through the floor slab at the operating floor level. This tie reduces the torsional effect on the beam, which is designed to resist the remaining torsional effects. Lateral stability of the columns is provided by reinforced concrete beams and slabs connected to the secondary shield wall at Elevations 0' and 25'.

The crane specification requires that the manufacturer's design insures that the crane will remain on the rail during either an OBE or an SSE. This is accomplished by means of kick-back plates attached to the wheel carriage, which prevent separation between the wheel and the rail. See Figure 3.8-30.

Seismic base shears are transmitted to the floor slab by means of a positive connection between the rail and the floor slab, which is part of the supporting structure. The crane rail anchor is designed to transfer all shear forces and overturning moments (resulting from the action of either an OBE or an SSE on the crane) to the concrete supporting system. Details of this connection include clamp plates, which restrain the rail and are welded to the base plate, and anchor bolts, which secure the base plate to the concrete beam. See Figure 3.8-30.

m. Neutron Shielding

The neutron shielding is a shell structure mounted on the slabs and walls above and around the reactor vessel. It consists of a plastic material encased in steel plate and provides radiation shielding.

A more complete description of the neutron shielding and its effect on structures can be found in Section 6.2.

3.8.3.2 Applicable Codes, Standards and Specifications

The design, materials, fabrication and inspection of the internal structures conform to, but are not necessarily limited to, the applicable sections of the following codes and specifications which are used to establish or implement design bases and methods, analytical techniques, material properties and quality control provisions. Dates and revisions given for the listed codes are the earliest version that was used. Subsequent issues were incorporated into the design where practicable or where the new issue directly affected the safety of the structure.

Code or Specification	Title
ACI 211.1-70	Recommended Practice for Selecting Proportions for Normal Weight Concrete
ACI 214-65	Recommended Practice for Evaluation of Compression Test Results of Field Concrete
ACI 301-72	Specification for Structural Concrete for Building (application sections)
ACI 302-69	Recommended Practice for Concrete Floor and Slab Construction
ACI 304-73	Recommended Practice for Measuring, Mixing, Transporting and Placing Concrete
ACI Committee Report 74-33	Recommended Practice for Hot Weather Concreting
ACI Report 306R-78	Recommended Practice for Cold Weather Concreting
ACI 308-71	Recommended Practice for Curing Concrete
ACI 309-72	Recommended Practice for Consolidation of Concrete
ACI Committee	Placing Concrete by Pumping Method
ACI Committee Report 72-33	Placing Concrete with Belt Conveyors
ACI 311-64	Recommended Practice for Concrete Inspections

Code or Specification	Title
ACI 315-65	Manual of Standard Practice for Detailing of Reinforced Concrete Structures
ACI 318-71*	Building Code Requirements for Reinforced Concrete (with Commentary)
ACI 347-68	Recommended Practice for Concrete Formwork
ACI 614-59	Recommended Practice for Measuring, Mixing, and Placing Concrete
ACI SP2 (1975 Edition)	ACI Manual of Concrete Inspection
CRSI	Reinforced Concrete - Manual of Standard Practice, 22nd Edition, first printing, 1976
CRSI	Recommended Practice for Placing Reinforcing Bars, 1968
ASME	ASME Boiler and Pressure Vessel Code, Section II, Material Specification, Part C - Welding Rods, Electrodes and Filler Metals (up to and including Winter 1974 Addenda)
ASME	ASME Boiler and Pressure Vessel Code, Section III, Division 2 - Code for Concrete Reactor Vessels and Containments (1975 Edition)
ASME	ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF (up to and including Winter 1974 Addenda)
ASME	ASME Boiler and Pressure Vessel Code, Section V, Nondestructive Examination (up to and including Winter 1974 Addenda)
ASME	ASME Boiler and Pressure Vessel Code, Section VIII, Pressure Vessels, Part 1 (up to and including Winter 1974 Addenda)

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* Since the internal structures are reinforced and not prestressed, cannot be classified as thin shells or special ductile frames (no formation of plastic hinges allowed from seismic loads) and are not designed as special shear walls, the provisions of ACI 318-71 code Chapters 18, 19 and Appendix A are not applicable.

Code or Specification	Title
ASME	ASME Boiler and Pressure Vessel Code, Section IX, Welding and Brazing Qualifications (up to and including Winter 1974 Addenda)
ASTM A1-68a	Standard Specifications for Carbon Steel Rails
ASTM A6-70	Specifications for General Requirements for Delivery of Rolled Steel Plates, Shapes, Sheet Piling and Bars for Structural Use
ASTM A53-73	Standard Specifications for Welded and Seamless Steel Pipe
ASTM A108-73	Standard Specifications for Cold Finished Carbon Steel Bars and Shafting
ASTM A123-73	Standard Specification for Zinc (Hot Galvanized) Coatings on Products Fabricated from Rolled, Pressed and Forged Steel Shapes, Plates, Bars and Strip
ASTM A143-74	Recommended Practice for Safeguarding Against Embrittlement of Hot-Dip Galvanized Structural Steel Products and Procedure for Detecting Embrittlement
ASTM A153-73	Standard Specification for Zinc Coating (Hot-Dip) on Iron and Steel Hardware
ASTM A185-70	Specification for Welded Steel Wire Fabric for Concrete Reinforcement
ASTM A193-75	Specification for Alloy Steel and Stainless Steel Bolting Material for High Temperature Service
ASTM A194-74	Standard Specification for Carbon and Alloy Steel Nuts and Bolts for High Pressure and High Temperature Service
ASTM A240-71	Specification for Stainless and Heat Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet and Strip for Fusion-Welded Unfired Pressure Vessels

Code or Specification	Title
ASTM A276-75	Standard Specification for Stainless and Heat Resisting Bars and Shapes
ASTM A307-68	Specification for Low Carbon Steel Externally and Internally Threaded Standard Fasteners
ASTM A312-74	Standard Specification for Seamless and Welded Austenitic Stainless Steel Pipes
ASTM A325-71	Specification for High Strength Bolts for Structural Steel Joints Including Suitable Nuts and Plain Temperature Service
ASTM A358-75	Standard Specification for Electric-Fusion-Welded Austenitic Chromium-Nickel Alloy Steel Pipe for High Temperature Service
ASTM A370-75a	Methods and Definitions for Mechanical Testing of Steel Products
ASTM A384-72	Safeguarding Against Warpage and Distortion During Hot-Dip Galvanizing of Steel Assemblies
ASTM A385-62	Recommended Practice for Providing High Quality Coatings (Hot Dip) on Assembled Products
ASTM A386-73	Standard Specification for Zinc Coatings (Hot Dip) on Assembled Products
ASTM A391-65	Standard Specification for Alloy Steel Chains
ASTM A394-75	Specification for Galvanized Steel Transmission Tower Bolts and Nuts
ASTM A446-72	Standard Specification for Steel Sheet Zinc Coated (Galvanized) by the Hot-Dip Process Physical (Structural) Quality
ASTM A479-75	Standard Specification for Stainless and Heat-Resisting Steel Bars and Shapes for Use in Boilers and Other Pressure Vessels
ASTM A480-74	Standard Specification for Delivery of Flat-Rolled Stainless and Heat Resisting Steel Plate, Sheet and Strip

Code or Specification	Title
ASTM A490-71	Specification for Quenched and Tempered Alloy Steel Bolts for Structural Steel Joints
ASTM A501-74	Standard Specification for Hot-Formed Welded and Seamless Carbon Steel Structural Tubing
ASTM A502-75	Standard Specification for Structural Steel Rivets
ASTM A514-70	Specification for High-Yield Strength, Quenched and Tempered Alloy Steel Plate, Suitable for Welding
ASTM A516-74a	Specification for Pressure Vessel Plates, Carbon Steel, for Moderate and Lower Temperature Service
ASTM A525-73	Standard Specification for General Requirements for Delivery of Steel Sheet, Zinc Control (Galvanized) by the Hot-Dip Process
ASTM A540-70	Alloy Steel Bolting Materials for Special Applications
ASTM A570-72	Hot-Rolled Carbon Steel Sheet and Strip
ASTM 572-75	High Strength Low Alloy, Columbium-Vanadium Steels of Structural Quality
ASTM A588-77a	Specification for High-Strength Low Alloy Structural Steel with 50,000 psi Minimum Yield Point to 4 in. Thick
ASTM A615-72	Specification for Deformed Billet-Steel Bars for Concrete Reinforcement
ASTM A759-78	Standard Specification for Carbon Steel Crane Rails
ASTM C29-71	Standard Methods of Test for Unit Weight of Aggregate
ASTM C31-69	Standard Method of Making and Curing Concrete Compressive and Flexural Strength Test Specimens in the Field
ASTM C33-71a	Specification for Concrete Aggregates
ASTM C39-71	Standard Method of Test for Compressive Strength of Molded Concrete Cylinders

Code or Specification	Title
ASTM C40-73	Standard Method of Test for Organic Impurities in Sands for Concrete
ASTM C42-68	Standard Method of Obtaining and Testing Drilled Cores and Sawed Beams of Concrete
ASTM C70-73	Standard Method of Test for Surface Moisture in Fine Aggregate
ASTM C87-69	Standard Method of Test for Effect of Organic Impurities in Fine Aggregate on Strength of Mortar
ASTM C88-73	Standard Method of Test for Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate
ASTM C94-72	Specification for Ready-Mixed Concrete
ASTM C109-73	Standard Method of Test for Compressive Strength of Hydraulic Cement Mortars (Using 2-inch Cube Specimens)
ASTM C114-69	Standard Methods of Chemical Analysis of Hydraulic Cement
ASTM C117-69	Standard Method of Test for Materials Finer than No. 200 (75 M) Sieve in Mineral Aggregates by Washing
ASTM C123-69	Standard Method of Test for Light Weight Pieces of Aggregate
ASTM C125-74	Standard Definitions of Terms Relating to Concrete and Concrete Aggregates
ASTM C127-73	Standard Method of Test for Specific Gravity and Absorption of Coarse Aggregates
ASTM C128-73	Standard Method of Test for Specific Gravity and Absorption of Fine Aggregate
ASTM C131-69	Standard Method of Test for Resistance to Abrasions of Small Size Coarse Aggregate by Use of Los Angeles Machine
ASTM C136-71	Standard Method of Test for Sieve or Screen Analysis of Fine and Coarse Aggregate

Code or Specification	Title
ASTM C138-75	Standard Method of Test for Unit Weight, Yield and Air Content (Gravimetric) of Concrete
ASTM C142-71	Standard Method of Test for Clay Lumps and Friable Particles in Aggregates
ASTM C143-71	Standard Method of Test for Slump of Portland Cement Concrete
ASTM C150-71	Specification for Portland Cement
ASTM C151-74a	Standard Method of Test for Auto-clave Expansion of Portland Cement
ASTM C172-71	Standard Method of Sampling Fresh Concrete
ASTM C173-75	Standard Method of Test for Air Content of Freshly Mixed Concrete by the Volumetric Method
ASTM C186-73	Standard Method of Test for Heat of Hydration of Hydraulic Cement
ASTM C191-74	Standard Method of Test for Time of Setting of Hydraulic Cement by Vicat Needle
ASTM C192-69	Standard Method of Making and Curing Concrete Test Specimens in the Laboratory
ASTM C231-71T	Tentative Method of Test for Air Content of Freshly Mixed Concrete by the Pressure Method
ASTM C233-73	Standard Method of Testing Air-Entraining Admixtures for Concrete
ASTM C235-68	Standard Method of Test for Scratch Hardness of Coarse Aggregate Particles
ASTM C260-69	Specification for Air-Entraining Admixtures for Concrete

Code or Specification	Title
ASTM C289-71	Standard Method of Test for Potential Reactivity of Aggregates (Chemical Method)
ASTM C294-69	Description of Nomenclature of Constituents of Natural Mineral Aggregates
ASTM C295-65	Recommended Practice for Petrographic Examination of Aggregates for Concrete
ASTM C309-74	Standard Specification for Liquid Membrane-Forming Compounds for Curing Concrete
ASTM C404-70	Standard Specification for Aggregates for Masonry Grout
ASTM C494-71	Specification for Chemical Admixtures for Concrete
ASTM C496-71	Standard Method of Test for Splitting Tensile Strength of Molded Concrete Cylinders
ASTM C535-69	Standard Method of Test for Resistance to Abrasion of Large Size Coarse Aggregate by Use of the Los Angeles Machine
ASTM C566-67	Standard Method of Test for Total Moisture Content of Aggregate by Drying
ASTM C666-75	Standard Method of Test for Resistance of Concrete to Rapid Freezing and Thawing
ASTM D75-71	Standard Method of Sampling Aggregates
ASTM D512-67	Standard Methods of Test for Chloride Ion in Industrial Water and Industrial Waste Water
ASTM D1411-69	Standard Method of Test for Water-Soluble Chloride Present as Admixtures in Graded Aggregate Road Mixes
ASTM D1888-67	Standard Method of Test for Particulate and Dissolved Matter in Industrial Water
ASTM E109-63	Standard Method for Dry Powder Magnetic Particle Inspection

Code or Specification	Title
ASTM E329-72	Recommended Practice for Inspection and Testing Agencies for Concrete as Used in Construction. (Articles 7, 8 and 9 do not apply.)
AISC	Specification for the Design, Fabrication and Erection of Structural Steel for Building 1969 Edition (including Supplements 1, 2 and 3)
ANS 20.1 (ANSI N177)	Proposed Standard for the Design Basis for the Protection Against Internal and External Plant Missiles (April 1974)
ANSI A58.1-1972	American Standard Building Code Requirements for Minimum Design Loads in Buildings and Other Structures
ANSI B31.1-1973	Summer and Winter Addenda, Power Piping
ANSI N45.2-1974	Quality Assurance Program Requirements for Nuclear Power Plants
ANSI N101.2-1972	Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities
ANSI N101.4-1972	Quality Assurance for Protective Coatings Applied to Nuclear Facilities
ANSI N101.6-1972 *	American National Standard for Concrete Radiation Shield
ANSI N512-1974	Protective Coating (Paints) for Nuclear Industry
AWS B3.0-41	Standard Qualification Procedures
AWS D1.0-69	Standard Code for Arc and Gas Welding in Building Construction

* Exception is taken to those sections of ANSI N101.6 that are not applicable to nuclear power plants. The applicable sections of ANSI N101.6 are those listed in Regulatory Guide 1.69.

Code or Specification	Title
AWS D1.1-75	Structural Welding Code
AWS D1.1-75	Recommended Practice for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction
American Railway Engineering Association (AREA)	Manual of Railway Engineering, Volumes I & II (1972 Revision)
American Hot Dip Galvanizing, Inc.	Hot Dip Galvanized Coatings (1973)
ASNT SNT-TC-1A (June, 1975)	American Society for Non-Destructive Testing Recommended Practice for Non-Destructive Testing-Personnel Qualifications and Certification
Bethlehem Steel	Crane Rails Catalog No. 3351 (updated to December 1979)
Bethlehem Steel	Specification for Fully Heat Treated Rails (March 1977 Revision)
Mixer Manufacturers Bureau of the Associated General Contractors of America	Concrete Plant Mixer Standards (1974 Issue)
MMB of the AGCA	Recommended Guide Specification for Batching Equipment and Control Systems in Concrete Batch Plants Publication 102 (1974 Issue)
NRMCA	National Ready-Mixed Concrete Association Certification of Ready-Mixed Concrete Production Facilities Instructions and Check List (1972)
NRMCA	Truck Mixer and Agitator Standards of Truck Mixers Manufacturer's Bureau (1971)
U.S. Army Corps of Engineers Specification CRD-C119-63	Method of Test for Flat and Elongated Particles in Coarse Aggregate

Code or Specification	Title
U.S. Dept. of Labor OSHA	Occupational Safety and Health Administration Standards, 1975 Edition
U.S. Dept. of Commerce National Bureau of Standards	PS-1-74 Construction and Industrial Plywood
U.S. Dept. of Commerce National Bureau of Standards	Handbook 44, Specifications, Tolerances and Other Technical Requirements for Commercial Weighing and Measuring Devices, 1971
SSPC	Steel Structures Painting Council Steel Structures Painting Manual, Volume 2; Systems and Specifications including supplement, 1973
SSPC	Good Painting Practice, Volume 1, 1966
Uniform Building Code	International Conference of Building Officials, Uniform Building Code, 1973 Edition
NRC TID 7024	Nuclear Reactors and Earthquakes, (August 1973)
NRC 10 CFR 50 App. B	Quality Assurance Criteria for Nuclear Power Plants
Nuclear Construction Issues Group (NCIG-01, Rev. 2, 5/7/85)	Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants

NRC Regulatory Guide No.	Title
1.10, Rev. 1, 1/73	Mechanical (Cadweld) Splices in Reinforcing Bars of Concrete Containments
1.15, Rev. 1, 12/72	Testing of Reinforcing Bars for Concrete Structures
1.29, Rev. 3, 9/78	Seismic Design Classification
1.54, Rev. 0, 8/72	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
1.55, Rev. 0, 6/73	Concrete Placement in Category I Structures
1.60, Rev. 1, 12/73	Design Response Spectra for Seismic Design of Nuclear Power Plants
1.61, Rev. 0, 10/73	Damping Values for Seismic Design of Nuclear Power Plants
1.69, Rev. 0, 12/73	Concrete Radiation Shields for Nuclear Power Plants
1.82, Rev. 0, 6/74	Sumps for Emergency Core Cooling and Containment Spray Systems Revision 3 was used for guidance for the replacement sump strainers along with NEI 04-07.
1.92, Rev. 1, 2/76	Combining Modal Responses and Spatial Components in Seismic Response Analysis
1.94, Rev. 1, 4/76	Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During Construction Phase of Nuclear Power Plants
1.142, Rev. 0, 4/78	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)

The below listed UE&C design and construction specifications applicable to the containment internal structures and to the other seismic Category I structures were prepared in accordance with applicable codes, quality control requirements, and NRC Regulatory Guides:

UE&C Specification	Title
9763.006-12-1	Structural Steel Work
9763.006-12-2	Structural Steel Erection
9763.006-12-5	Fabrication of Safety-Related Structural Steel Work
9763.006-13-3	Category I Concrete Work Other Than Containment
9763.006-14-1	Furnishing, Detailing, Fabricating, and Delivering Reinforcing Bars
9763.006-14-3	Installation of Reinforcing Bars in Category I Structures (Other Than Containment)
9763.006-18-1	Furnishing of Miscellaneous Embedded Steel and Weldments
9763.006-18-2	Installation of Miscellaneous Steel and Weldments
9763.006-18-3	Furnishing of Miscellaneous Steel
9763.006-18-4	Furnishing and Installing Embedded Steel and Miscellaneous Steel
9763.006-41-4	Furnishing of Protective Coating (Paint) System Materials and Related Services
9763.006-41-7	Field Painting of Containment Structure Interior
9763.006-69-1	Concrete Batch Plant
9763.006-69-3	Concrete Mixes
9763.006-69-6	Ready-Mixed Concrete from Off-Site Plant
9763.006-69-7	Standard Concrete Mixes
9763.006-69-10	Ready-Mixed Concrete for Category I, Non-Category I Structures and Systems from an Off-Site Plant

UE&C Specification	Title
9763.006-225-2	Stainless Steel Liners
9763-MPS-1	Material and Processing Requirements (Nuclear)
9763-MPS-2	Material and Processing Requirements (Non-Nuclear)
9763-MPS-3	Material and Processing Requirements of Welded Studs, Reinforcing Bars and Anchor Bolts
9763-QAS-1	Administrative and System Requirements (Nuclear)
9763-QAS-2	Administrative and System Requirements (Non-Nuclear)
9763-RM-1	Instructions for Site Records Management System
9763-WS-2	Requirements for Welding and Non-Destructive Examination for Non-Nuclear Pressure Components and Non-Nuclear Power Piping
9763-WS-3	Requirements for Welding and Non-Destructive Examination for Structural Steel
9763-WS-4C	Requirements for Cadwelding and Non-Destructive Examination of Mechanical Rebar Splice Method

3.8.3.3 Loads and Loading Combinations

The containment internal structures are designed to withstand all credible conditions of loading including normal loads, severe environmental loads, extreme environmental loads, and abnormal plant loads. These loads are determined in accordance with the applicable specifications, including ACI 318-71 and AISC-69, and are considered in Normal and Unusual Load Combinations to assure that the response of the structure remains within the limits prescribed in Subsection 3.8.3.5. Refer to Subsection 1.8 (Regulatory Guide 1.142) for a discussion concerning compliance with ACI-349.

Seismic Category I structures with concrete affected by ASR shall meet the acceptance criteria of the Codes of Record, with the following list of deviations which are considered as supplements to the Codes of Record.

Supplement Number	Description Reference NextEra Energy Seabrook FP# 101196
1	Consideration of ASR loads: The UFSAR load and load combinations Tables 3.8-1, 3.8-14, and 3.8-16 were modified in LAR 16-03 to consider the ASR load and load factors for calculating the total demands on structures affected by ASR.
2	Code acceptance criteria: Strength of reinforced concrete sections affected by ASR can be calculated using the Codes of Record (ASME 1975 and ACI 318-71) and the minimum specified design concrete strength, provided that ASR expansion is within the limits provided in Table 3.8-18 for through-thickness and volumetric expansion.
3	Shear-friction capacity for members subjected to net compression: The shear-friction capacity for members subjected to net compression can be calculated using procedures defined in Building Code Requirements for Reinforced Concrete (ACI 318-83 Section 11. 7).
4	Flexural Cracked Section Properties: Reductions of the gross cross-sectional moment of inertia for analysis shall be computed considering the presence of cracking and the prestressing effects of ASR; alternatively, 50% of the gross cross-sectional moment of inertia can be used.
5	Axial and Shear Cracked Section Properties: Axial and shear cracking reduces the corresponding stiffness of a structural member. The effect of cracking on reducing the axial and shear stiffness of structural components may be considered in analysis.

a. <u>Design Loads</u>

The definitions of the loads used in the design of the internal structures include the following:

1. <u>Normal Plant Startup, Operation and Shutdown Loads</u>

Normal loads are those loads encountered during normal plant start-up, operation and shutdown. They include the following:

(a) Dead Loads (D)

Dead loads are all permanent gravity loads including the weight of concrete walls and slabs, structural framing, piping, cable and cable trays, permanent equipment, and static pressures of liquids. Concrete creep, shrinkage, and swelling are considered for structures affected by the expansion of concrete from alkali-silica reaction (ASR). See Subsection 3.8.4.6 for a description of the effects of ASR on concrete.

(b) Live Loads (L)

Live loads include any movable equipment loads and other loads which vary in intensity and/or occurrence. Live loads are present only during shutdown conditions, and do not govern the design of any components.

(c) Operational Thermal Loads (T_o)

The temperature gradient through the walls under normal operating conditions is considered in the design. For a discussion of minimum and maximum operating temperatures, see Subsection 6.2.1.

(d) Operational Pipe Reactions (R_o)

These are pipe reactions due to thermal conditions existing in the piping during normal operation or shutdown. They are based on the most critical transient or steady-state condition.

(e) Alkali-Silica Reaction Loads

These are structural effects caused by ASR. These loads are considered if ASR is observed or identified in concrete structures internal to Containment. ASR loads are passive and therefore occur during normal operation, shutdown conditions, and concurrently with all extreme environmental loads. For Containment internal structures, only the effects of ASR expansion occurring in reinforced concrete structural members are considered. (Expansion of concrete backfill is not considered as the concrete backfill does not interact with Containment internal structures.) Calculation of these demands is described below; detailed guidance on calculation of the loads is provided in "Methodology for the Analysis of Seismic Category I Structures with Concrete Affected by Alkali-Silica Reaction," (FP# 101196).

Demands associated with internal ASR expansion shall be applied to structural components as strain loads based on in-plane expansion measurements. The demands associated with internal ASR expansion shall be applied uniformly through the cross-sectional thickness of the structural components (e.g., walls, slabs, foundations, etc.) unless otherwise justified.

Large in-plane expansion measurement values may not necessarily imply large ASR expansions. If some or all of the cracks at an ASR monitoring grid are shown to be caused by a mechanism other than internal ASR in the reinforced concrete member (e.g., shrinkage, thermal, pressurization tests of Containment, structural cracks due to external loads, etc.), then the in-plane expansion measurements should be adjusted accordingly. The adjusted in-plane expansion value shall be computed by excluding the widths of cracks determined not to be caused by ASR.

2. Severe Environmental Loads

Severe environmental loads are those loads that result from events that could infrequently be encountered during the plant life. The only load included in this category is the following:

(a) Operating Basis Earthquake (E_o)

These are the loads generated by the Operating Basis Earthquake, which is the earthquake that could reasonably be expected to affect the site during the operating life of the plant. Only the actual dead load and weights of fixed equipment are considered in evaluating the seismic response forces.

The horizontal and vertical design response spectra for the OBE are derived by applying a factor of 0.5 to the response spectra given for the Safe Shutdown Earthquake (SSE) which is described below. The effects of two (2) orthogonal horizontal components and one (1) vertical component of earthquake are considered and combined by the square-root-of-the-sum-of-the-squares rule.

3. Extreme Environmental Loads

Extreme environmental loads are those loads which result from events which are credible but highly improbable. The only load included in this category is the following:

(a) Safe Shutdown Earthquake (E_{ss})

These are the loads generated by the Safe Shutdown Earthquake, which is the earthquake based upon an evaluation of the maximum earthquake potential in the vicinity of the plant. Dead and fixed equipment loads are described under the Operating Basis Earthquake, above. The horizontal and vertical forces on the internals are developed from the response spectra given in Figure 2.5-38 and Figure 2.5-39 the development of which is described in Subsection 2.5.2.6.

The effects of two (2) orthogonal horizontal earthquakes and one (1) vertical earthquake are considered and combined by the square-root-of-the-sum-of-the-squares rule.

4. Abnormal Loads

Abnormal loads are those loads generated by a postulated high energy pipe break resulting in a loss-of-coolant accident (LOCA). The design basis accident results in the highest postulated pressures and temperatures, and is determined by considering a LOCA.

(a) Accident Pressure (P_a)

Immediately following a postulated primary pipe break and prior to pressure equalization, a differential pressure occurs on the interior structures. See Subsection 6.2.1.2 for a description of these pressure loads, including the effects of the neutron shielding on pressures. Note that the differential pressure load on the primary shield wall is greater than the differential pressure load for the secondary shield wall; however, in this subsection one symbol is used to represent this loading with the understanding that the value is as specified in Subsection 6.2.1.2. Accident pressures resulting from a main steam line break are also considered.

(b) Accident Temperature (T_a)

Also following a postulated primary pipe break is an increase in temperature on the interior structures. These loads are also further described in Subsection 6.2.1.3.

(c) Accident Pipe Reactions (R_a)

Pipe reaction loads due to thermal conditions generated by the postulated pipe break, including R_o, are considered in the design. The magnitude of these loads is determined by the piping design.

(d) Pipe Break Loads (R_r)

Pipes, other than the Primary Coolant System, are anchored in sleeves through the interior concrete structures and transmit thrust loads to the structure during normal operating conditions and for the postulated pipe break. In addition, loads transmitted by pipe whip restraints are considered. The three components of pipe break loads are as follows:

(1) R_{rr} = load on the structure generated by the reaction of a ruptured high energy pipe during the postulated pipe break. The time-dependent nature of the load and the ability of the containment to deform beyond yield are considered in establishing the structural capacity necessary to resist the effects of R_{rr}.

(2) R_{rj} = load on the structure generated by jet impingement from a ruptured high energy pipe during the postulated pipe break. In general, direct impingement of steam on the structure does not produce significant design loadings due to the distance between the wall and the break location. Where a break is postulated to occur close enough to a wall to produce a critical loading, a shield, or deflector, is provided, and the loading is transferred to the embedment for the pipe whip restraint to which the shield is attached.

(3) R_{rm} = load on the containment resulting from the impact of a ruptured high energy pipe during the postulated pipe break. Since all high energy lines are constrained by pipe rupture restraints, missile loadings of this nature are prevented.

5. Other Loads and Load Considerations

(a) Internal Missile Loads (M)

Internal missile loads, other than those defined as R_{rr}, R_{rj}, and R_{rm}, are considered including an appropriate dynamic factor to account for the dynamic nature of the load. For a discussion of specific missiles generated inside the containment and the missile design procedures, see Section 3.5. Note that the sides and floor of the refueling canal are designed for the impact of the neutron shielding.

(b) Primary Equipment Supports

The primary equipment supports transmit loads to the fill mat, to the primary and secondary shield walls and to the operating floor slab. For a discussion of these loads and of the design criteria for the supports see Subsection 5.4.14.

(c) Load Considerations for Internal Structures

(1) Time dependent loads such as thermal effects, creep and shrinkage do not have any significant effects on the internal structures since the accident loads are generally resisted in tension by the reinforcing steel. In addition, the accident loads are short-term, once-occurring loads, which have negligible creep effects.

(2) The containment structure functions as an independent structure with complete physical separation from the internal structures, and therefore there are no loading interactions between the two.

(3) Compartmentalization is considered in the design of the internal structures by using the peak subcompartment differential pressures, plus a safety margin. This is further discussed in Subsection 6.2.1.2.

(4) Self-Straining Loads
The internal concrete structures are analyzed for deformation caused by ASR and are designed to withstand the effects of ASR expansion, creep, shrinkage, and swelling.

b. Load Combinations

Various load combinations are considered in design to determine the greatest strength requirements of the structure. Where varying loads occur, the combinations producing the most critical loading are used. Basic combinations in the design of the containment internal structures are given in Table 3.8-14. These load combinations are in agreement with Subsections II.3 and II.5 of the Standard Review Plan for Subsection 3.8.3 of the UFSAR. The factors which are to be applied to allowable stresses have been transposed and applied as load factors instead, resulting in acceptance criteria as indicated in the table. Two categories of loading conditions and criteria are used in the design of the containment internal structures as described below.

1. Normal Load Conditions

   Normal load conditions e those encountered during testing and normal operation. They include dead load, live load, Alkali Silica Reaction (ASR) loads, and anticipated transients or test conditions during normal and emergency startup and shutdown of the Nuclear Steam Supply, Safety and Auxiliary Systems. Normal loading also includes the effect of an Operating Basis Earthquake. Normal load conditions are referred to in Division 2 as service load conditions.

   Under each of these loading conditions, the structure is designed so that the behavior of structures is in the small deformation elastic range. Design assumptions are presented in Subsection 3.8.3.4 and stress limitations are presented in Subsection 3.8.3.5.

2. Unusual Load Conditions

   Unusual load conditions are those conditions resulting from combinations of the LOCA, SSE and OBE, high-energy pipe failures, and live and dead loads. They are referred to in Division 2 as factored load conditions.

   For each of the unusual loading combinations, the internal structures are designed to remain below their ultimate capacity so that the behavior of structural components is in the small deformation elastic range. Design assumptions are presented in Subsection 3.8.3.4 and stress limitations are presented in Subsection 3.8.3.5.

3.8.3.4 Design and Analysis Procedures

a. Design and Analysis

After preliminary design, the internal structures are analyzed to determine the maximum stress and displacements in reinforcing steel, concrete and structural steel for the loading criteria described in Subsection 3.8.3.3. Each structural system, either an individual component or a group of connected components, is analyzed separately using appropriate design assumptions and boundary conditions.

The analyses include fixed base conditions, uncracked concrete, load superposition, modeling and selective interpretation of results. Material properties and boundary conditions are selected to be representative of the particular structure or component being analyzed. Either conventional methods and formulae of structural mechanics or the finite element direct stiffness method are used. All analyses consider elastic behavior only. The design is based on codes, specifications and documents listed in Subsection 3.8.3.2.

The methods of design and analysis, including assumptions, boundary conditions, loads resisted and analysis techniques are given below. Seismic analysis is discussed in Subsection 3.7.2.

Reinforcing in concrete components is proportioned in accordance with the responses obtained from the analyses using the strength design method of the ACI 318-71 code, as applicable, and the requirements of Subsections 3.8.3.3 and 3.8.3.5. Bond and anchorage requirements are in accordance with ACI 318-71. As a result, reinforcing patterns in walls and floors, in general, consist of orthogonal layers on both the inside and outside faces of the concrete.

The computer program CONCOL (described in Subsection 3.8.3.4) was used throughout the internals for ultimate strength design of concrete. Other computer programs used in the design and analysis of specific internal structures are described below.

Structural steel is designed in accordance with the AISC Specification, using elastic methods, except as noted below for the reactor support system.

Refer to Subsection 1.8 (Regulatory Guide 1.142) for a discussion concerning compliance with ACI-349.

b. Internal Structures

1. Reactor Support System

Design and analysis of the vertical supports are described in Subsection 5.4.14.

The reactor vessel lateral support, the ring girder, is analyzed using a three-dimensional finite element model. Since the support possesses bilateral symmetry and is subjected only to unilateral horizontal loading, only half of the support was required for the model. Loadings used in the analyses were supplied by the NSSS vendor, Westinghouse.

Design and analysis procedures for the reactor vessel lateral support are in accordance with the ASME Boiler & Pressure Vessel Code, Division 1, Subsection NF and Appendix XVII.

2. <u>Steam Generator Support System</u>

Design and analysis of the steel supports are described in Subsection 5.4.14.

3. <u>Reactor Coolant Pump Supports</u>

Design and analysis of the reactor coolant pump supports are described in Subsection 5.4.14.

4. <u>Primary Shield Wall</u>

The reactor pressure vessel is enclosed and supported by a reinforced concrete primary shield wall. The primary shield wall is designed under the accident condition considering the effects of temperature, jet forces and forces transmitted by the reactor vessel, seismic loads, etc. The design pressure loads are conservatively considered as peak transient pressure differential on the wall. Other LOCA forces on the structure are determined using dynamic time-dependent analysis. The peak values of these loads are conservatively used as equivalent static loads.

The wall is analyzed as an axisymmetric structure resting on the fill mat using the WILSON 2 computer code (described in Subsection 3.8.1.4). The analysis is performed assuming base fixity and top restraint provided by the floor of the refueling canal. Differential pressure loads and thermal loads are analyzed as axisymmetric load cases. Differential pressures are considered as nonaxisymmetric loads and are represented by a Fourier series.

An additional consideration is the effect of radiation in generating heat in the primary shield wall. This wall is the only concrete subjected to relatively high irradiation. The attenuation of the integrated neutron and gamma rays within the wall will produce a temperature rise which will be limited to a maximum of 150°F at the concrete surface adjacent to the reactor vessel by the convection of cooling air circulated between the wall and the reactor vessel.

Studies on the behavior of concrete under irradiation (Reference 5) indicate that irradiated specimens behave in a manner similar to control specimens subjected to temperature without radiation. Thus, the effects on the shield wall concrete from heat generated by irradiation were evaluated in the same manner as temperature rise occurring in the concrete from a nonradiating source.

5. Secondary Shield Wall

The secondary shield wall is analyzed as part of a larger three-dimensional model which includes the secondary shield wall, the refueling canal, the operating floor slab, the crane support structure and the biological shield wall. The computer program used was STARDYNE (described in Subsection 3.8.1.4).

The model is composed of plate elements (which include bending and transverse shear) for the walls and slabs and beams for the crane support structure and other uses as required. Restraint is provided at the level of the fill mat and at the interfaces between the operating floor and the primary shield wall. Differential pressures, thermal loads, pipe break loads, and normal operating pipe loads are applied to the finite element model. Force and displacement boundary conditions imposed by the primary shield wall onto the operating floor slab are considered.

6. Refueling Canal

The refueling canal is analyzed as part of the model described above under the heading, "Secondary Shield Wall."

7. Pressurizer Region

The pressurizer region is analyzed for the established load criteria using an independent three dimensional finite element model and the general procedures described above under the heading, "Secondary Shield Wall."

8. Control Rod Drive Mechanism (CRDM) Missile Shield

The CRDM missile shield has been integrated into the simplified head assembly. The missile shield has been evaluated to the criteria presented in the Standard Review Plan (SRP) Section 3.5.3. The missile shield and supporting structure have been modeled using the ANSYS computer code.

9. Operating Floor Slab

The operating floor slab is analyzed as part of the model described above under the heading, "Secondary Shield Wall."

10. Fill Mat

The fill mat is analyzed as an independent circular reinforced concrete slab; the model consists of plate elements (which include bending and transverse shear). It is designed to resist and transfer to the containment base mat all reaction loads from the internal structures including the primary equipment supports. The internal structures are generally rigidly connected to the fill mat.

11. Structural Steel

Structural steel framing and platforms, including connections, are conventionally designed using applicable codes and specifications, as listed in Subsection 3.8.3.2. Those members which support piping, electrical cable trays and ducts were further evaluated for rigidity considerations as described below.

All members of the containment annulus steel framing that support piping or piping and electrical cable trays and/or ducts were initially designed to have natural frequencies of 20 hz or greater, in order to minimize the seismic loads in the supported components. For members supporting only ducts and/or electrical cable trays the minimum frequency was 15 hz.

Where meeting these criteria is not practicable, members supporting piping were initially designed for a minimum frequency of 15 hz, if practicable.

Members that were not sized using the frequency approach were initially designed for static loads and an additional 2.0 g factor (horizontal and vertical).

Final member sizes were all checked for conformance with applicable codes against final loadings obtained from dynamic analyses of steel framing, piping, ducts and equipment.

12. Crane Support Structure

    The crane support structure is analyzed as part of the model described above under the heading, "Secondary Shield Wall."

13. Control Rod Drive Mechanism (CRDM) and Drive Rod Position Indication (DRPI) Cable Bridges

    To support the CRDM and DRPI cables that run to the simplified head assembly, two hinged cable bridges have been mounted on the operating deck, one on each side of the refueling cavity. The bridges have been modeled using the STAAD III computer code.

c. Description of Computer Programs Utilized in the Design and Analysis

The computer programs used in the design and analysis of the containment internal structures are described briefly in this subsection. Programs listed below are only those programs which have not been described previously in Subsections 3.8.1.4g and 3.8.2.4i. A summary of the comparisons of results used to validate them is given in Appendix 3F. The program classifications are discussed in Subsection 3.8.1.4g.

1. CONCOL (SYSTEM PROFESSIONAL): Documentation is available from Control Data Corporation. CONCOL is a computer program for ultimate strength design of concrete columns.

2. STRAP: "Static Analysis of Linear Elastic Structures," by United Engineers & Constructors, Inc. STRAP is used to perform static analyses of structures which can be represented as an assemblage of linear elastic members. It uses the stiffeners matrix methods of analysis.

3. GENSAP (STRU-PAK): "Static Analysis of Elastic Structures." Documentation is available from Control Data Corporation. GENSAP is used for general static analysis of elastic structures composed of beams and columns.

4. STAAD III Structural Analysis and Design Software: A proprietary computer program of Research Engineers, Inc. (REI), California, for the analysis and design of structures. The code has been placed under configuration control by Westinghouse and specific features of this software utilized in the simplified head assembly calculations have been independently verified by Westinghouse. (Westinghouse Letter Number EDRE-CSE-134(97), Software Release of STAAD III (22.0W) on Windows NT System, 9/25/97, and Westinghouse Calculation #CSE-06-98-0001, Rev. 0, titled: "STAAD Verification Problem-Response Spectra Analysis.")

5. GOTHIC Generation of Thermal-Hydraulic Information for Containments: GOTHIC is a general purpose thermal-hydraulic computer program for design, licensing, safety and operating analysis of nuclear power plant containments and other confinement buildings. GOTHIC has been developed for the Electric Power Research Institute (EPRI) by Numerical Applications Inc, Richland, Washington. The code has been placed under configuration control by Westinghouse. (GOTHIC Qualification Report for Version 5.0e (NAI 8907-09, Rev. 3, Dec. 1995.)

3.8.3.5 Structural Acceptance Criteria

The bases for the development of the following stress-strain criteria are the ACI 318-71 and AISC codes.

a. Normal Load Conditions

Internal structures are proportioned to maintain elastic behavior under all normal loading conditions described in Subsection 3.8.3.3.

Reinforced Concrete - designed in accordance with ACI 318-71 Strength Method, which insures flexural ductility by control of reinforcing steel percentages and stresses.

Structural and Miscellaneous Steel - designed in accordance with AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, Part I.

b. Unusual Load Conditions

Internal structures are designed to maintain elastic behavior under all unusual load conditions shown in Subsection 3.8.3.3. The upper bound of elastic behavior is taken as the yield strength capacity of the load carrying components. The yield strength of structural and reinforcing steel is taken as the minimum guaranteed yield stress as given in the appropriate ASTM Specification.

Reinforced Concrete - designed in accordance with ACI 318-71 Building Code. Member yield strength is considered to be the strength capacity calculated by the ACI Code.

Structural and Miscellaneous Steel - designed in accordance with AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, Part 1.

Overall stability of steel structures designed for unusual loading is verified using the AISC Specification, Part 2, and the load factors in Table 3.8-14, with the exception of the reactor vessel lateral support, which is designed in accordance with ASME B&PV Code, Division 1, Subsection NF.

c. Deformations

Each of the structures is designed to remain in the small deformation elastic range so that no gross deformations will occur and cause contact with other structures or pieces of equipment.

d. Seismic Analysis

The seismic analysis techniques used for seismic Category I structures and systems are described in Subsection 3.7.2. The seismic analysis techniques used for seismic Category I subsystems are described in Subsection 3.7.3.

The shear produced by the seismic response of the internal structures is accounted for in design. If the available shear strength of the concrete section is not adequate, shear reinforcement is used to increase the shear resistance of the section. In general, seismic loads are not the governing factor in the design of the internal structures. Consequently, shear produced by seismic response is resisted by orthogonal layers of reinforcing provided for the applicable design loading combination; diagonal seismic reinforcing is not required. The shear response of the internal structures is transferred to the fill mat which is keyed into the containment base mat at the reactor cavity pit.

3.8.3.6 Materials, Quality Control and Special Construction Techniques

The primary materials of construction for the internal structures are concrete, reinforcing steel, structural steel (rolled shapes and plates), stainless steel (liner plate), and field coatings. Descriptions of these materials and respective quality control are discussed below. There are no special construction techniques.

a. Concrete

Concrete work is in accordance with ACI 318-71 and ACI 301 codes, except as noted in Subsection 1.8 (Regulatory Guide 1.55). The concrete is a dense, durable mixture of sound, coarse aggregate, cement and water. Admixtures were added, where required, to improve the quality and workability during placement and to retard the set of the concrete. Engineering approval was required prior to the use of admixtures.

Aggregate conforms to ASTM C33. It consists of inert materials that are clean, hard and durable, free from organic material and uncoated with clay or dirt. Fine aggregate consists of natural sand and the coarse aggregate consists of crushed stone.

Portland cement conforms to ASTM C150, Type II (moderate heat of hydration requirements).

Water is clean and free from any deleterious amounts of acid, alkali, salts, oil, sediment, organic matter or other substances which may be harmful to the concrete or steel.

The reinforced concrete has a nominal density of 150 lb/ft^3, which is used for determination of dead load. Shielding calculations for the primary shield wall are based on a dry concrete density of 139 lb/ft^3; other shielding calculations are based on a dry concrete density of 147 lb/ft^3. The 28-day standard compressive strength of the concrete is 4000 psi.

Refer to Subsection 3.8.4.6 for a discussion on the material effects of Alkali Silica Reaction (ASR) on concrete.

To assure that adequate means of control were used in the manufacture and that the properties described above were realized, the following were required:

1. Suppliers, fabricators and contractors were required to have written quality assurance procedures, which were reviewed and approved by United Engineers. Material certifications were required in accordance with the applicable portions of the quality assurance plan described in Chapter 17 of the UFSAR and in the material specifications.

2. The following tests and certifications were required:

    (a) Aggregates were tested to comply with ASTM C33.

    (b) Cement was tested in accordance with ASTM C114 to conform to ASTM C150.

    (c) Concrete samples were taken from the mix in accordance with ASTM C172.

    (d) Cylinders were made in accordance with ASTM C31.

    (e) Compressive tests were made in accordance with ASTM C39.

    (f) Slump tests were in accordance with ASTM C143 and air content tests were in accordance with ASTM C231.

    (g) Evaluation of the tests was in accordance with the material specifications, the codes discussed above, and the ACI 318-71 code, as applicable.

3. All making and testing of concrete samples was accomplished by an independent testing laboratory.

    All concrete operations during cold weather conditions followed the practice defined in ACI 301 and 306R-78.

    During cold weather curing of the concrete, concrete surfaces whose temperatures are below 50°F by accident for short periods of time, but remain 40°F or above, have had the 7-day curing period extended by the amount of time the concrete was below 50°F (rounded out to the nearest whole day).

The strength maturity method as defined in ACI 306R-78 was allowed as an alternate to specified curing requirements. The required maturity factor was selected to insure that concrete attained 70 percent of the design strength. This method was also used to determine the attainment of in-place concrete strength required for removal of forms in lieu of field cured cylinder as required in ACI 301.

b. Reinforcing Steel

Reinforcing steel consists of high-strength deformed billet bars conforming to ASTM A615, Grade 60. Certified Material Test Reports and user tests, as required by Regulatory Guide 1.15, were provided by the material manufacturer.

All reinforcing bars were spliced in accordance with UE&C Specification 9763-WS-4C, and No. 14 and 18 bars were joined by mechanical butt splices (Cadweld splices).

Additional information on requirements for reinforcing steel can be found in Subsection 3.8.1.6.

c. Structural Steel

All structural and miscellaneous steel, including stainless, conforms to the following specifications unless otherwise noted.

Structural steel conforming to ASTM A36 was used except where high strength is required. ASTM A588 or A572, Grade 50 was used for these applications, such as the reactor vessel lateral support, the cross-over leg restraint and the hot leg restraint.

To assure that steel plates, such as embedded plates for pipe restraints, are able to transmit orthogonal loads, a material with good through-plate tensile properties, ASTM A36 steel, was used.

Bolts were made of ASTM A325 steel except in special applications. Where high strength was required ASTM A193, A490, or A540 bolts were used. ASTM A307 bolts were used for stair stringers, stair treads, grating, handrail, toe plates, girts and purlins.

Stainless steel liner plate was fabricated from material conforming to ASTM A240, TP-304 steel. A liquid penetrant test is performed on the liner plate on the refueling canal to assure leak tightness.

The crane rail is a 171 lb/ft rail fabricated from A-759 steel; it was fully heat treated.

All welding conforms to either the American Welding Society Code (AWS) or to the ASME Boiler and Pressure Vessel Code (Section III).

For structural steel governed by AWS D1.1 and not subject to fatigue loads, welds shall be visually inspected in accordance with NCIG-01, "Visual Weld Acceptance Criteria (VWAC)," as specified in UE&C Specification WS-3. The only exception is for structural steel governed by AWS D1.1 which are subject to fatigue; no undercut is permitted.

Welders of carbon steel are qualified in accordance with the "Standard Qualification Procedure" of the AWS; welders of stainless steel liner plate are qualified in accordance with Section IX of the ASME Boiler and Pressure Vessel Code.

Certified Material Test Reports giving chemical composition and physical properties were supplied by the manufacturers for all structural steel.

Fabrication and erection was in accordance with AISC standards and the applicable material specifications.

d. Steel and Concrete Coating System

Materials used for coating the internal structures are the same as those described in Subsection 3.8.1.6. These materials meet the requirements of ANSI Standard N101.2.

3.8.3.7 Testing and In-Service Surveillance Requirements

Quality control testing as discussed in Subsection 3.8.3.6 will be employed. No additional testing or in-service surveillance is required.

Refer to Subsection 3.8.4.7 for a discussion on the Structural Monitoring Program requirements for inspection of ASR affected structures

3.8.4 Other Seismic Category I Structures

This section contains physical descriptions, codes, loads and load combinations, design and analysis procedures, allowable stresses, quality control, and testing requirements as they relate to seismic Category I structures exclusive of the containment structure and its internals. Nonseismic Category I structures which received special design considerations to prevent collapse onto adjacent Category I structures are also included in this section.

3.8.4.1 Description of the Structures

Locations of all of the major plant structures are shown on the plant arrangement plan, Figure 1.2-1; overall dimensions of the seismic Category I structures are given in Table 3.8-15.

All seismic Category I structures are separated from adjacent structures above the point of fixity by means of isolation joints, except for a few structures described below in which two portions of the same structure are used for two different functions. The resulting multi-function structure is isolated from adjacent structures. See Subsection 3.8.5.1 for a detailed discussion of seismic isolation.

The seismic Category I structures are described in the following paragraphs. They incorporate no unique or new design or construction features. Also, block or concrete masonry partitions are not utilized in any Category I structure.

a. Containment Enclosure Building

The Enclosure Building is a reinforced concrete right cylindrical structure with a hemispherical dome. The inside diameter of the cylinder is 158 feet. The vertical wall varies in thickness from 36 inches to 15 inches; the dome is 15 inches thick. The inside of the dome is 5'-6" above the top of the containment dome.

Located on the outside of the Enclosure Building is the plant vent stack, consisting of a light steel frame with steel plates and varying in cross section. The stack carries exhaust air from various buildings.

The space between the containment and the Enclosure Building is maintained at a slight negative pressure during accident conditions. All joints and penetrations are caulked or gasketed to ensure air tightness. See Figure 1.2-2, Figure 1.2-3, Figure 1.2-4, Figure 1.2-5 and Figure 1.2-6 for sections and elevations of the Enclosure Building.

b. Containment Equipment Hatch Missile Shield

This shield is a removable, precast, reinforced concrete wall located outside the equipment hatch. It serves to protect the hatch from tornado-generated missiles. See Figure 1.2-4 and Figure 1.2-5 for a plan and a section.

c. Containment Enclosure Ventilation Area

The containment enclosure ventilation area is an irregularly shaped reinforced concrete building that houses ventilation equipment (fans, filters, etc.) for the Enclosure Building and is located on the southwest side of the containment. Its overall dimensions are 116.0 feet long, 55.75 feet wide at its widest point, and 31.5 feet high. See Figure 1.2-2, Figure 1.2-3, and Figure 1.2-4.

d. Control and Diesel Generator Building

The Control and Diesel Generator Building is a reinforced concrete structure approximately 233 feet long by 90 feet wide. All load carrying walls and columns are founded on fill concrete and rock below grade. This is a multi-function structure in which the two portions, the control room area and the diesel generator area, are separated by a common wall in the north-south direction and are not seismically isolated. The building was analyzed and designed as a unit.

The east portion of the structure (Control Building), which is 138 feet long, has three floors and extends from grade to approximately 79 feet above grade. The two intermediate floors and roof are supported on steel columns in the center and on concrete walls all around. The ground floor contains electrical switchgear, motor generator sets and battery rooms; the second floor is for cable spreading; and the third floor is the main control room.

The west portion of the structure (Diesel Generator Building), which is 95 feet long, has two floors and extends from 36 feet below grade to approximately 59 feet above grade. The portion below grade houses storage tanks for diesel fuel. The area between elevations 20'-0" and 50'-0" is divided, north and south, by a 2 feet thick reinforced concrete wall which supports the second floor and provides protection for each diesel generator against missiles generated by the other. The second floor contains air intakes for the diesel generators and building ventilation equipment. The roof is supported by concrete walls all around and by steel columns in the center extending from the second floor and located directly over the dividing missile wall below.

See Figure 1.2-31, Figure 1.2-32, and Figure 1.2-33, Figure 1.2-34, Figure 1.2-35 and Figure 1.2-36 for plans and sections of the Control Building and Diesel Generator Building, respectively.

e. Control Room Makeup Air Intake Structure

The control room makeup air intake structures are reinforced concrete structures which serve as terminals for buried ductwork that provides air for the control rooms during accident conditions. The intake is located near the demineralized water tank, north of the Circulating Water Pumphouse.

The top of the sleeve for the air intake is located at Elevation 22'-7", higher than the maximum flood elevation.

See Figure 3.8-31 for a plan and section.

f. Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Areas (Control Building to Containment)

The Emergency Feedwater Pump Building is a reinforced concrete building on the north end of the Enclosure Building. It extends to approximately 28 feet north of the Enclosure Building and is 84 feet wide. In elevation, the building extends from Elevation (-)26'-0" (top of floor slab) to elevation 47'-0" (top of roof slab). The building consists of the emergency feedwater pump room located above a two-story high electrical cable tray tunnel. The roof is supported on three exterior walls and on columns, on the south side. Portions of the emergency feedwater pump rooms which extend beyond the electrical cable tray tunnels are supported on walls founded on rock or on concrete fill.

The emergency feedwater pump room contains emergency feedwater pumps, demineralized water makeup pumps, valve stations and an auxiliary control panel. A monorail is provided for servicing the pumps. The electrical penetration areas are approximately 84 feet wide and are situated one on top of the other. They penetrate the Enclosure Building and join with the containment structure.

See Figure 1.2-2, Figure 1.2-3, Figure 1.2-4, and Figure 1.2-6, and Figure 1.2-51.

g. Enclosure for Condensate Storage Tank

The enclosure is a cylindrical reinforced concrete wall, 2 feet thick and 43 feet high, which surrounds each tank and extends up to the springline of the tank, providing protection from horizontal tornado-generated missiles. The enclosure is capable of retaining the contents of the tank should the tank rupture due to a missile penetrating it from above. See Figure 3.8-32.

h. Fuel Storage Building

The Fuel Storage Building is a reinforced concrete structure approximately 98 feet square that extends approximately 44 feet below grade to 66 feet above grade. The building contains the new fuel storage area and the spent fuel pool. The spent fuel pool is constructed of concrete walls of a minimum thickness of 6 feet and a 4'-10" thick concrete floor; for leak tightness the inside surface is lined with stainless steel plates 3/16" thick on the walls and 1/4" thick on the floor. The building superstructure is constructed of concrete walls and roof.

New and spent fuel storage and handling are described in Section 9.1. Refer to Figure 1.2-15, Figure 1.2-16, Figure 1.2-17, Figure 1.2-18, Figure 1.2-19, Figure 1.2-20 and Figure 1.2-21 for plan and sectional elevations of the building.

i. Main Steam and Feedwater Pipe Chase (East) Including East Penetration Area

The main steam and feedwater pipe chase (east) is a reinforced concrete structure which houses and protects the main steam and feedwater piping. The overall dimensions are 127.58 feet long, 22.25 feet wide for most of the length, and 61.5 feet high. It is 41.25 feet wide at the widest point.

The east penetration area is a room located at the southern end of the pipe chase which houses the control panels for the hydrogen recombiner. See Figure 1.2-3 and Figure 1.2-4.

j. Main Steam and Feedwater Pipe Chase (West) Including Mechanical Penetration Area and Personnel Hatch Area

The main steam and feedwater pipe chase (west) is a reinforced concrete structure which houses and protects the main steam and feedwater piping. The pipe chase is 113.75 feet long, 20.0 feet wide for most of the length, and 61.5 feet high (overall); the pipe chase area is 59 feet high. It is 23.75 feet wide at the widest point.

Located below the chase area is the mechanical penetration area which houses piping running between the Containment and the Primary Auxiliary Building. This region is partitioned into several smaller areas which include the radiation and nonradiation shield tunnels. The mechanical penetration area is 78.0 feet long, 34.71 feet wide at the widest point, and 37.5 feet high.

The personnel hatch area is an irregularly shaped reinforced concrete structure located outside the personnel hatch of the containment, for which it provides protection from missiles and illegal entry. It is connected to the west pipe chase and has approximate overall dimensions of 46.75 feet length, 37.25 feet width, and 29.25 feet height. See Figure 1.2-2, Figure 1.2-3, Figure 1.2-4, and Figure 1.2-5.

k. Piping Tunnels

The piping tunnels are underground reinforced concrete structures, rectangular in section, which are located throughout the plant. They house the piping systems found in the plant. See Figure 3.8-33.

l. Pre-Action Valve Building

The Pre-Action Valve Building is an irregularly shaped reinforced concrete building which contains the deluge valve for the Fire Protection System. It is approximately 34.83 feet long, 34.75 feet wide and 27.33 feet high in overall dimensions and is located on the east side of the Emergency Feedwater Building. See Figure 1.2-51.

m. Primary Auxiliary Building Including Residual Heat Removal (RHR) Equipment Vault

The Primary Auxiliary Building is a reinforced concrete structure. The major portion is approximately 79 feet wide by 145 feet long and extends from 13 feet and 46 feet below grade to 88 feet above grade. The equipment vault is attached to the structure and is 57 feet by 43 feet. It extends from 81 feet below grade to 5.5 feet above grade. The entire building is founded on rock or concrete fill.

The RHR equipment vault is subdivided into six compartments by continuous concrete walls as follows: two for containment spray pumps and heat exchangers, two for residual heat removal pumps and heat exchangers and two for access stairs. Plugs are provided in the reinforced concrete roof for removal of the heat exchangers.

Located on the east and north sides of the RHR equipment vault are a personnel walkway and an electrical cable tray chase located above it. This structure is supported on fill concrete over rock and connected to the RHR equipment vault at its base, Elevation 20'-8"; it extends to Elevation 53'-0". The east portion is approximately 13.0 feet wide and 41.0 feet long; the north portion is approximately 10.0 feet wide and 38.0 feet long.

That portion of the building which is 79 feet by 145 feet, the Primary Auxiliary Building, has two intermediate reinforced concrete floors which support miscellaneous auxiliary nuclear equipment, such as heat exchangers, pumps, demineralizers, filters, tanks and ventilation equipment. Reinforced concrete walls and steel columns support the intermediate floors and reinforced concrete roof slab.

Below-grade reinforced concrete pipe tunnels connect the building to the Containment and Waste Processing Building. Monorail hoists are provided to handle materials and servicing of equipment.

See Figure 1.2-9, Figure 1.2-10, Figure 1.2-11, and Figure 1.2-12 for plans and sectional elevations of the Primary Auxiliary Building and Figure 1.2-13 and Figure 1.2-14 for plans and sectional elevations of the RHR equipment vault, including the personnel walkway and electrical cable tray chase.

n. Safety-Related Electrical Duct Banks and Manholes

Safety-related electrical duct banks are reinforced concrete structures, rectangular in cross section, which enclose cables running between various seismic Category I buildings other than the containment. Cross-sectional dimensions vary from duct to duct and lengths are as required by the separation of the buildings being connected. The safety-related electrical duct banks are completely below grade and principally supported on engineered fill. They are isolated from the buildings and manholes by flexible connections at the connection points.

Direction changes of the duct banks between buildings are accomplished by intermediate reinforced concrete manholes, which are generally rectangular in cross section. Overall dimensions vary depending on the size and number of duct banks entering the manhole.

See Figure 3.8-34 and Figure 3.8-35 for the duct banks and manholes respectively.

o. Service Water Cooling Tower Including Switchgear Rooms

The Service Water Cooling Tower is a rectangular building approximately 300 feet x 54 feet in plan, extending 28 feet below grade, and projecting 57.5 feet above ground. It is located on the south side of the plant and is founded on rock. The cooling tower serves as an ultimate heat sink in the unlikely event that the cooling water tunnels are rendered inoperative. The cooling tower houses pumps, fans, water distribution system and nozzles.

The switchgear rooms are approximately 54 feet x 26 feet in plan and extend from Elevation 22.0 feet to Elevation 46.0 feet. They are located on the east and west ends of the cooling tower and are an integral part of the structure. The switchgear rooms house the switchgear, substation, and motor control center for the cooling tower. See Figure 1.2-56.

Operational characteristics of the cooling tower are described in Subsection 9.2.5.

p. Service Water Pumphouse

The Service Water Pumphouse (SWPH) is located on the plant site east of the Containment Building and is founded on rock. It is attached to and shares a foundation with the Circulating Water Pumphouse (CWPH), a non-Category I structure which is designed so that its loss or failure will not impair the Service Water Pumphouse or System.

The structure under the SWPH is a reinforced concrete basin and is approximately 91 feet wide by 74 feet long; it extends from the operating floor, 1 foot above grade, to 63 feet below grade. The Service Water Pumphouse itself is approximately 118 feet wide by 78 feet long and extends 28 feet above the operating floor to Elevation 49'-0". The walls, roof and interior columns are reinforced concrete.

A service water electrical control room on the west end of the SWPH is a reinforced concrete structure founded on rock and is integrally connected to the SWPH. It is approximately 48 feet wide by 38 feet long and extends to 19.5 feet above grade.

The Circulating Water Pumphouse is approximately 119 feet wide by 123 feet long and extends 28 feet above the operating floor to Elevation 49'-0". The building is non-Category I and consists of a steel frame covered with metal siding, from the grade to the roof level, and roofing. A non-Category I trash removal room on the north end is a steel framed structure with metal siding and is approximately 39 feet wide by 17 feet long extending to 14 feet above grade to Elevation 34'-0".

The structure under the CWPH is a reinforced concrete basin and is approximately 110 feet wide by 123 feet long; it extends from the operating floor to 63 feet below grade.

The basin under the SWPH is integrally connected to the basin under the CWPH by a common east-west wall. This common basin, the SWPH and the electrical room are designed as a unit.

See Figure 1.2-46, Figure 1.2-47,and Figure 1.2-48 for plans and sectional elevations of the Service Water Pumphouse as well as the Circulating Water Pumphouse.

Operational characteristics of the Service Water System are described in Subsection 9.2.1, and operational characteristics of the Circulating Water System are described in Subsection 10.4.5.

q. Tank Farm (Tunnels) Including Dikes and Foundation for Refueling Water Storage Tank (RWST) and Reactor Makeup Water Storage Tank (RMUWST)

The tank farm area consists of a reinforced concrete portion and structural steel framing portion. The reinforced concrete portion including the foundation, dike walls, pipe tunnels and pipe chases are structures associated with safety-related systems and are designed as Seismic Category I.

Structural steel framing portion which includes steel framing, concrete roofing and metal siding, is used to enclose the area above the tanks and to form the motor control center and switchgear room. The steel framing portion is designated nonseismic Category I and designed and constructed so that the safe shutdown earthquake (SSE) would not cause the steel framing portion to collapse upon any safety-related structure, system or component within or surrounding the tank farm area.

The tunnels provide a passageway for piping which runs between the Primary Auxiliary Building and either the Service Water Pumphouse or the Service Water Cooling Tower.

The dikes are reinforced concrete walls surrounding the tanks and extending to Elevations 42'-0" and 30'-0" for the RWST dike and the RMUWST dike, respectively. Structural steel framing is used between the tops of the dikes and the roof framing, which is also structural steel.

Overall dimensions are 152 feet in length, approximately 66 feet in width and 92 feet in height. See Figure 1.2-23, Figure 1.2-24, Figure 1.2-25 and Figure 1.2-27.

r. Waste Processing Building

The Waste Processing Building is approximately 108 feet wide by 189 feet long and extends 51 feet below grade to 91 feet above grade.

The building consists of a reinforced concrete portion and a structural steel portion with a 4" nominal concrete slab.

The reinforced concrete portion between column lines 1 to 2 and A to D, housing radioactive gaseous waste equipment and carbon delay bed, is designed as seismic Category I. To maintain integrity of this portion, the entire structure is designed as seismic Category I, except for siding, girts and roofing on steel at El. 111'-0. However, local effects of seismic loads from system supports is not accounted for, since they do not affect overall safety of the structure.

The safety-related portion of the concrete between column lines 1 to 2 and A to D is designed to withstand tornado loads. The structural main frame above El. 53'-0" is designed to withstand tornado wind loads.

The building contains liquid and radioactive gaseous waste processing and solid waste systems. A trucking facility is provided along the south wall for shipping drums and containers.

See Figure 1.2-22, Figure 1.2-23, Figure 1.2-24, Figure 1.2-25, Figure 1.2-26, Figure 1.2-27, Figure 1.2-28, Figure 1.2-29 and Figure 1.2-30 for plans and sectional elevations of the Waste Processing Building.

s. <u>Nonseismic Category I Structures</u>

Several nonseismic Category I structures were designed against collapse onto seismic Category I structures due to tornado wind and SSE loadings. These structures can be found on the site arrangement plan, Figure 1.2-1, and include the following:

- Turbine Building
- Nonessential Switchgear Building
- Tank Farm Area (Steel Framing Portion)
- Steam Generator Blowdown Recovery Building

The Turbine Building is described in Subsection 1.2.2, and the tank farm area (steel framing portion) is described earlier in this subsection.

The Nonessential Switchgear Building is located on the north side of the Control Building and houses and protects the electrical equipment used to provide lighting for the plant.

The Steam Generator Blowdown Recovery Building is located on the south side of the Waste Processing Building and tank farm area, and houses the Steam Generator Recovery System.

Specific design measures taken to protect the effected seismic Category I structures are given in Subsections 3.3.2 and 3.7.2 for tornado wind and SSE loadings, respectively.

t. Service Water Access Vault

The Service Water Access Vault (SWAV) is located underground on the plant site north of the Cooling Towers. It is a reinforced-concrete structure approximately 14 feet wide by 38 feet long; it extends from 15 feet to 29 feet below grade. A precast concrete manway extends from the top of the vault to approximately 2 feet below grade. The vault is founded on fill concrete and rock. The SWAV provides access to the underground safety-related A and B Train 24" service water piping to permit entry into the piping for field weld joint refurbishment and joint inspection. See Figure 3.8-37.

3.8.4.2 Applicable Codes, Standards and Specifications

Codes, standards and specifications listed in Subsection 3.8.3.2, as well as the following US NRC regulatory guides, are also applicable to other seismic Category I structures:

Regulatory Guide No.	Title
1.76 (4/74)	Design Basis Tornado for Nuclear Power Plants
1.117 (4/78)	Tornado Design Classification

3.8.4.3 Loads and Loading Combinations

The seismic Category I structures discussed herein were designed, as applicable, for all credible conditions of loadings, including normal loads, loads due to severe and extreme environmental conditions, abnormal loads and missile loads. These loads were determined in accordance with the applicable codes, including ACI 318-71 and AISC-69, and were considered in normal and unusual load combinations to assure that the structures remain within the limits prescribed in Subsection 3.8.4.5.

a. Design Loads

The definitions of the loads used in the design of the other seismic Category I structure include the following:

1. Normal Startup, Operational, and Shutdown Loads

Normal loads are those loads encountered during normal plant operation, startup and shutdown. They include the following:

(a) Dead Loads (D)

Dead loads are all permanent gravity loads including, but not limited to, concrete walls and slabs, structural framing, piping, cable and cable trays, permanent equipment and miscellaneous building items. Hydrostatic pressures of liquids are also included in this category. Concrete creep, shrinkage, and swelling are considered for structures affected by the expansion of concrete from alkali-silica reaction (ASR). See Subsection 3.8.4.6 for a description of the effects of ASR on concrete.

(b) Live Loads (L)

Live loads are all temporary gravity loads including but not limited to normal snow loads, conventionally distributed and concentrated floor loads, and movable equipment loads, such as cranes and hoists.

Equipment operating loads and impact factors are the greater of those recommended by the manufacturer or the applicable building codes.

Unusual snow load (L_s), which is greater in magnitude than normal snow load, was also used where applicable. Lateral earth pressures due to soil backfill (H) were used where applicable. Three types of lateral earth pressure loading, active, at rest and passive, were considered, with pressures determined by acceptable theories of soil mechanics.

(c) Operational Thermal Loads (T_o)

These are the thermal effects and loads occurring during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

(d) Operational Pipe Reactions (R_o)

These are the pipe reactions occurring during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

(e) Alkali-Silica Reaction Loads

These are structural effects caused by ASR. ASR loads are passive and therefore occur during normal operation, shutdown conditions, and concurrently with all extreme environmental loads. The structural effects from ASR expansion include demands associated with internal ASR expansion of structural components, and, as applicable, the pressure exerted on embedded structures by ASR expansion of the concrete backfill. Calculation of these demands is described below; detailed guidance on calculation of the loads is provided in “Methodology for the Analysis of Seismic Category I Structures with Concrete Affected by Alkali-Silica Reaction,” (FP# 101196).

Internal ASR Expansion of Structural Components

Demands associated with internal ASR expansion shall be applied to structural components as strain loads based on in-plane expansion measurements. The demands associated with internal ASR expansion shall be applied uniformly through the cross-sectional thickness of the structural components (e.g., walls, slabs, foundations, etc.) unless otherwise justified.

Large in-plane expansion measurement values may not necessarily imply large ASR expansions. If some or all of the cracks at an ASR monitoring grid are shown to be caused by a mechanism other than internal ASR in the reinforced concrete member (e.g., shrinkage, thermal, structural cracks due to external loads, etc.), then the in-plane expansion measurements should be adjusted accordingly. The adjusted in-plane expansion value shall be computed by excluding the widths of cracks determined not to be caused by ASR.

External Load from ASR Expansion of Concrete Backfill

ASR expansion of concrete backfill can create an external pressure on the walls and slabs of structures and can lead to structural deformation, rigid body displacement of structures, and relative displacements between adjacent structures. ASR expansion of concrete backfill cannot easily be measured directly. External pressure on the walls and slabs due to ASR expansion of concrete backfill is determined indirectly through field measurements of structural displacements and deformations, and/or through the use of conservative assumptions, as described in FP# 101196. The magnitude of concrete backfill pressure can be adjusted by correlating the structural analysis deformation under in-situ conditions to field observations.

ASR expansion of concrete backfill is considered for structures embedded in concrete backfill without an isolation gap. This approach is conservative as shrinkage of the concrete backfill and structure and possible deterioration of the waterproofing backboard would result in separation.

2. <u>Severe Environmental Loads</u>

Severe environmental loads are those loads which could infrequently be encountered during the plant life. The following loads are included in this category:

(a) Operating Basis Earthquakes (E_o)

These are the loads generated by the operating basis earthquake, which is the earthquake of an intensity of the same level as the highest intensity that has been experienced historically at the site. Only the actual dead load and weights of fixed equipment were considered in evaluating the seismic response forces. The horizontal and vertical design response spectra for the OBE were derived by applying a factor of 0.5 to the response spectra given for the safe shutdown earthquake (SSE), which is described below. The effects of two (2) orthogonal horizontal components and one (1) vertical component of earthquake were considered and combined by the square-root-of-the-sum-of-the-squares method. Lateral dynamic soil loads, including hydrodynamic loadings, due to the OBE (H_e) were also used where applicable.

(b) Wind Load (W)

Wind loads generated by the design wind specified for the plant site were considered. See Subsection 3.3.1 for a discussion of wind velocity and applied forces.

3. Extreme Environmental Loads

Extreme environmental loads are those loads which result from postulated events which are credible, but highly improbable. The following loads are included in this category:

(a) Safe Shutdown Earthquake Loads (E_s)

These are the loads generated by the safe shutdown earthquake, which is the maximum potential earthquake that could occur in the vicinity of the plant, based on geological and historical investigations. Only the actual dead load and weights of fixed equipment were considered in evaluating the seismic response forces. The horizontal and vertical responses on the structures were developed from the response spectra given in Figure 2.5-38 and Figure 2.5-39, the development of which is described in Subsection 2.5.2.6. The effects of two (2) orthogonal earthquakes and one (1) vertical earthquake were considered and combined by the square-root-of-the-sum-of-the-squares method. Lateral dynamic soil loads, including hydrodynamic loadings, due to the SSE (H_s) were also used where applicable.

(b) Tornado Loads (W_t)

Wind loads generated by the design tornado specified for the plant site were considered along with the differential pressure loads due to rapid atmospheric pressure drop and the tornado-generated missile effects.

See Subsection 3.3.2 for a discussion of the design basis tornado and its associated wind and pressure loadings. See Section 3.5 for a discussion of tornado-generated missiles.

4. Abnormal Loads

Abnormal loads are those loads generated by postulated high energy pipe ruptures, particularly a rupture in the Reactor Coolant System resulting in a loss-of-coolant accident (LOCA).

(a) Accident Pressure (P_a)

This is the design pressure load occurring within a structure due to the DBA.

(b) Accident Temperature (T_a)

This includes the thermal effects and loads generated by the DBA including T_o.

(c) Accident Pipe Reactions (R_a)

Pipe reaction loads due to thermal conditions generated by the postulated pipe break, including R_o, were considered in the design.

(d) Pipe Break Loads (R_r)

These are local effects on the structures due to the postulated pipe break, as follows:

(1) R_{rr} = load on the structure generated by the reaction of a ruptured high energy pipe during the postulated pipe break. The time-dependent nature of the load and the ability of the structure to deform beyond yield were considered in establishing the structural capacity necessary to resist the effects of R_{rr}.

(2) R_{rj} = load on the structure generated by jet impingement of a ruptured high energy pipe during the postulated pipe break. In general, direct impingement of steam on a wall of a structure does not produce significant design loadings due to the distance between the wall and the break location. Where a break is postulated to occur close enough to a wall to produce a critical loading, a shield, or deflector, is provided, and the loading is transferred to the pipe whip restraint to which the shield is attached.

(3) R_{rm} = the load on the structure resulting from the impact of a ruptured high energy pipe during the postulated pipe break. Since all high energy lines are constrained by pipe restraints, loading of this nature is prevented.

5. Site-Related Loads

(a) Flood Loads (F)

These are the loads resulting from the design basis flood including uplift and wave runup. See Section 3.4. for further discussion.

(b) Missile Loads (M)

Internal missile loads, other than those defined as R_r, are considered including an appropriate dynamic load factor. See Section 3.5 for a detailed discussion of these missiles.

b. Load Combinations

Various load combinations were considered in design to determine the strength requirements of the structure. Where varying loads occur, the combinations producing the most critical loading were used. The basic combinations considered in the design of each seismic Category I structure are given in Table 3.8-16.

Two categories of loading conditions and criteria were used in the design of the seismic Category I structures other than the containment, as described below:

1. Normal Load Conditions

   Normal load conditions are those encountered during testing and normal operation and are referred to in the standard review plan as service load conditions. They include dead load, live load and anticipated transients, loads occurring during normal startup and shutdown, and loads occurring during emergency shutdown of the nuclear steam supply, safety and auxiliary systems. Normal loading also includes the effect of an operating basis earthquake and normal wind load. Under each of these loading combinations the structures were designed such that deformation will be small and the structure will respond elastically. Design and analysis procedures are presented in Subsection 3.8.4.4 and stress limitations are presented in Subsection 3.8.4.5.

2. Unusual Load Conditions

   Unusual load conditions are those resulting from combinations of accident, wind, tornado, earthquake, live and dead loads and are referred to in the standard review plan as factored load conditions.

   For these loading combinations, the structures were designed to remain below their ultimate yield capacity such that deformations will be small and structural components will respond elastically. Design and analysis procedures are presented in Subsection 3.8.4.4 and stress limitations are presented in Subsection 3.8.4.5.

c. Other Load Considerations

1. Creep

   Effects of concrete creep are negligible due to the low sustained concrete stresses associated with conventionally reinforced concrete structures and, therefore, were not a governing factor in design. Concrete creep is included with other self-straining loads in the design of structures with deformation caused by the effects of ASR expansion.

2. Stability

   The other seismic Category I structures were checked for overturning, sliding, and flotation using the load combinations of Table 3.8-16 with the exception that the coefficient for live load is zero. Buoyant forces were considered to decrease the dead load in computing both overturning and sliding.

3. Self-Straining Loads

Structures that are analyzed for deformation caused by ASR are designed to withstand the effects of ASR expansion, creep, shrinkage, and swelling.

3.8.4.4 Design and Analysis Procedures

a. Design and Analysis

Category I structures other than the containment are constructed of reinforced concrete with structural steel framing used to support vertical loading on floor slabs. Structural steel is also used to provide enclosure for some areas, as described in Subsection 3.8.4.1, and for other miscellaneous purposes. Reinforced concrete structures consist of a system of walls and slabs generally to provide a continuous, integral framing system. Vertical forces are transferred to the foundation mats through the walls and structural steel and reinforced concrete columns. Lateral forces are transferred to the foundation mats primarily by the action of shear walls; some load is also transferred by means of flexural action of the walls, all of which are rigidly attached at the mat.

The Containment Enclosure Building, due to its cylindrical and hemispherical shape and relative dimensions, was analyzed as a three-dimensional, thin-shell structure. Boundary conditions were consistent with the support on rock and the lateral restraint provided by backfill concrete placed against the structure. Internal resultant forces and moments were determined by integration of the appropriate shell stresses through the thickness. Critical transverse shear force was derived by considering the variations in bending moments across the surface, in conjunction with the applied hydrostatic load (which produces additional local shear not reflected in the finite element analysis). Reinforcing was subsequently designed for these internal forces.

Columns are designed to resist other lateral loads, such as pipe loads or building displacements, in addition to those forces transmitted to the columns at floor levels. Steel columns are generally pin-connected at the foundation mat, and reinforced concrete columns are rigidly attached. Structural steel framing for floor systems primarily consists of pin-connected framing with some members being continuous. Rigidity is provided by the box-like concrete walls and slabs.

Structures with deformation were analyzed by calculating the self-straining load associated with the combined effects of ASR-expansion, creep, shrinkage, and swelling as determined from measurements of the structure. ASR expansion loads were combined with other loads and the appropriate load factors from Table 3.8-16 were applied.

Several computer programs were used for static analysis and are described in Appendix 3F.

Table 3.8-17 contains a list of these programs and the respective structures for which they were used. The load combinations are given in Subsection 3.8.4.3.

The idealization of each structure depended on its geometric configuration and applied loading. Both two and three-dimensional analyses were performed. When the loads are considered to act principally in a given plane the floor diaphragms are rigid in their own planes; and when the rotations between adjacent frames are uncoupled, two-dimensional analyses were used. Three-dimensional analyses were used when other analyses were not suitable.

Detailed analyses of local areas of these structures, which either are subjected to local loads such as thermal hydrostatic and hydrodynamic loads, or have unusual geometry, such as numerous openings, were performed where required. Such areas included the north wall of the main steam and feedwater pipe chase (east) and the dividing wall in the Control and Diesel Generator Building. In these analyses, a representative area of a structure was isolated, and force and/or displacement boundary conditions, consistent with the overall behavior of the structure, were imposed on the substructure in addition to the appropriate loading.

Structural systems of the buildings were designed for dead load, live load, and lateral loads, such as those loads produced by wind, tornado, and earthquake. All structural elements were designed to resist the effects of internally generated missiles, where applicable. Tornado loads consist of applied pressure and missile impact, for which all seismic Category I structures are designed except as indicated below.

The safety-related electrical duct banks and manholes, the mechanical penetration area and the pipe tunnel were not designed for any tornado loads. The duct banks and manholes are covered by a nominal 5 feet of backfill, minimum, which protects them from tornado loads. The mechanical penetration area is located beneath the west pipe chase, and the pipe tunnel is covered by approximately 15 feet of backfill. The manhole covers, however, were designed for tornado-generated missiles.

The Service Water Cooling Tower and the Containment Enclosure Building were designed for tornado pressure but not for tornado-generated missiles. Since the circulating water tunnels will be in operation during a tornado instead of the cooling tower and since the containment was designed for tornado-generated missiles, these structures need not be designed for missiles.

The Pre-Action Valve Building is designed for tornado-generated missiles and tornado depressurization loading (Elevation 27'-0" slab). The elevation 27'-0" slab is a tornado depressurization barrier for safety-related equipment and components located in the electrical tunnels.

In the Waste Processing Building only those areas housing radioactive gaseous waste equipment were designed for tornado loads; the other areas do not contain safety-related equipment. Procedures by which the structures were checked for missile loads, tornado-generated as well as internally generated, are described in Section 3.5. Determination of pressures on structures due to tornado is described in Subsection 3.3.2. Pressure loadings from wind are described in Subsection 3.3.1.

The seismic analysis of seismic Category I structures is described in Subsection 3.7.2. All cranes in these structures are furnished with hold-down devices to ensure that they are not dislodged by earthquake forces. Monorails, by nature of their support mechanisms, cannot be dislodged by earthquake forces.

Using methods outlined in TID-7024, "Nuclear Reactors and Earthquakes," the effects of hydrodynamic forces were included in the seismic analyses of the Service Water Cooling Tower. Also using methods outlined in TID-7024, the weight of constrained water and sloshing effects of water in motion were included as equivalent static loads in the final design of the Service Water Cooling Tower, Service Water Pumphouse and Fuel Storage Building.

Reinforced concrete design of Category I structures was in accordance with the strength design procedures of the ACI 318-71 code, except as indicated in Subsection 3.8.4.5. Structural steel design was in accordance with the provisions of the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings (1969 Edition). Refer to Subsection 1.8 (Regulatory Guide 1.142) for a statement concerning compliance with ACI-349.

b. Material Properties

Material properties were selected from the normal range of values to produce a conservative design. See Subsection 3.8.1.4 for a detailed discussion of the influence of material properties on design and analysis.

Analyses and test of ASR-affected concrete concluded that the capacity of structural members and embedded concrete anchors in ASR-affected structures is not reduced when ASR expansion levels are below the limits included in Subsection 3.8.4.7.

c. Computer Programs

The computer programs used in analysis and design of other Category I structures are listed in Table 3.8-17. These programs are further described in Appendix 3F.

3.8.4.5 Structural Acceptance Criteria

The basis for the acceptance criteria is the ACI 318 and AISC Codes. However, under the action of seismic or wind loadings, in accordance with the standard review plan (Section II.5), the 33 percent increase in allowable stresses was not permitted.

a. Normal Load Conditions

Structures were proportioned to maintain elastic behavior under all normal loading conditions described in Subsection 3.8.4.3. Reinforced concrete structures were designed in accordance with ACI-318 Strength Method, which insures flexural ductility by limiting reinforcing steel percentages and stresses.

Structural and miscellaneous steels were designed in accordance with AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, Part 1.

b. Unusual Load Conditions

Structures were proportioned to maintain elastic behavior under all unusual load conditions shown in Subsection 3.8.4.3. The upper bound of elastic behavior was taken as the yield strength capacity of the load carrying components. The yield strength of structural and reinforcing steel was taken as the minimum guaranteed yield stress as given in the appropriate ASTM Specifications. Reinforced concrete structures were designed in accordance with ACI-318 Building Code. Member yield strength was considered to be the strength capacity calculated by the ACI Code.

Structural and miscellaneous steels were designed in accordance with Part 1 of AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings.

c. Deformations

Since each of the structures was designed to be in the small deformation, elastic range, no gross deformations will occur that will cause significant contact with other structures or pieces of equipment. All deformations, however, were evaluated considering the relationship of the subject component to both adjacent and supporting structures and equipment.

In December 2014, an area of the Containment Enclosure Building (CEB) was identified as being deformed. AR02014325 Root Cause Evaluation (RCE) was completed which determined the cause of the deformation to be internal expansion (strain) in the CEB concrete produced by Alkali Silica Reaction (ASR) and ASR expansion in the backfill concrete adjacent to the below grade portions of CEB external walls. Deformation in the CEB resulted in contact and distortion of components in structures adjacent to the CEB. CEB deformation also resulted in reduction in the seismic isolation between the CEB and adjacent structures.

Operability determinations were performed which established reasonable assurances that the structure, and SSCs impacted by deformation of the structure, were operable and capable of performing their intended design function.

Degraded conditions of SSCs attributed to deformation of the structure that were not corrected are currently being monitored under Seabrook's Corrective Action Program. Any further degradation will initiate corrective actions to ensure operability of the affected SSCs is maintained.

The operability determinations will remain open, and monitoring noted above will continue, until a License Amendment has been approved that allows NextEra Energy to evaluate and monitor ASR induced deformation in Seabrook Station structures.

d. Additional Stress-Strain Considerations

Stress-strain limits and design parameters were applied to the design of each applicable element for the specific values identified with each loading combination and design condition identified in Subsection 3.8.4.3.

No special allowance has been made for variation of material properties over the life of the structure, beyond that which is taken into account in establishing allowable stresses, strains, capacity reduction factors, concrete protection of reinforcing, and crack control as outlined in the referenced ACI and AISC codes. Additional corrosion protection is provided to concrete structures by means of waterproofing for parts of the structure below grade and by painting, coating or installing of liners for structural concrete tanks (such as the spent fuel pool in the Fuel Storage Building).

Variations in stress and strain, due to scheduled plant shutdowns and startups, have negligible effect on the overall structural behavior because of the small variation in the average structure temperature and loading. Since the designs of the structures were governed by extreme, infrequently occurring loadings, such as tornadoes and earthquakes, and normal cyclical changes in stress levels are comparatively small, no reduction in the margin of safety will occur over the life of the plant.

All connections and joints were designed to transfer all design forces (shear, tension and compression) and moments with a safety margin and degree of conservatism that is required by the applicable code.

e. Stability

Acceptance criteria for stability are given in Subsection 3.8.5.5.

3.8.4.6 Materials, Quality Control and Special Construction Techniques

The primary materials of construction are concrete, reinforcing steel and structural steel (rolled shapes and plates).

Descriptions of the materials and basic quality control procedures are discussed in Subsection 3.8.3.6.

Alkali-aggregate reactions (AAR) occur over time in hardened concrete between the alkali hydroxides in the pore solution of concrete and certain minerals found in some aggregates. Alkali Silica Reaction (ASR) is the predominant type of AAR. It involves a chemical reaction between alkalis in the cement paste (Portland cement) and reactive forms of silica in the aggregates. This reaction is dependent on several factors including the amount and form of reactive material in the aggregate (e.g., reactive forms of quartz), the amount of alkali in the cement (more alkali - faster reaction), temperature (higher temperature - higher reaction rate), and moisture content. The reaction forms an expansive gel in the affected material. As the reaction progresses and the gels expand, micro-cracks are formed in the aggregate extending into the cement paste. The main observable effect of ASR affected structures is expansion and cracking due to gel formation. As expansions increase, visible cracks begin to form on the exposed surfaces. These cracks are often in a characteristic pattern cracking and may also have signs of ASR gel material. While very reactive aggregates can cause rapid expansion rates that manifest in visible cracks and measurable expansion rates in a few years, ASTM testing for reactive aggregates and specification of low alkali cement have been somewhat effective in preventing ASR in these time frames. Slow reacting aggregates may not manifest for decades. The concrete constituents used at Seabrook would not be expected to be susceptible to ASR since:

1. the coarse aggregate is largely igneous rock that was routinely tested during construction and passed petrographic examinations and expansive reaction tests that normally detect alkali-silica reaction; and
2. low-alkali Portland cement was abundantly used.

Aggregates routinely passed ASTM reactivity and expansion tests per C227 and C289. Petrographic examinations of aggregates per C295 were performed but did not detect presence of reactive aggregates. In retrospect, the testing standards in place at the time of original construction may not accurately predict late or slow reactive aggregates. Empirical evidence at Seabrook suggests the coarse aggregates are the slow reactive type.

In June 2010, concrete cores were removed for examination and testing from the walls of the lower electrical tunnel in the Control Building, as part of preparations for license renewal inspections. The purpose was to evaluate potential concrete aging effects in below grade areas of the plant that had been subjected to historical groundwater wetting of the concrete. In general the removed cores showed the expected quality materials and placements from original construction. There were no obvious visual signs of aging distress or concrete degradation. Petrographic examinations were performed which involved sectioning and polishing the core samples and analysis under magnification by a qualified professional petrographer. This analysis revealed micro cracks and other features indicative of Alkali Silica Reaction (ASR). Materials testing of the removed cores also resulted in lower than expected mechanical properties consistent with low levels of ASR. The impact of ASR in the material strength testing of removed cores is not indicative of actual in situ performance and cannot be directly correlated to actual structural impact. Once removed from the structural context (e.g., reinforcement or confining loads) the behavior of the cores no longer reflects that of the confined structure.

The expansion of concrete from silica gel formation and cracking results in a small, but measurable, change in the dimensions of ASR-affected concrete. In 2014, Engineering determined that damage to a vertical seismic gap seal between the Containment Enclosure Building and the Containment Building was caused by relative building movement and not seal degradation. A subsequent evaluation of the condition determined that the seal damage was caused by radial deformation of the Containment Enclosure Building. The building deformation was caused by the expansion of ASR-affected concrete in the building and the backfill concrete that abuts the structure below grade.

Engineering evaluations of the extent of deformation in each structure determined the impact of the ASR on affected structures. Subsection 3.8.4.7 and the Structural Monitoring Program includes criteria for monitoring the effects of ASR.

Additional concrete core sampling has been performed to determine the extent of condition both from the perspective of additional areas that might be affected by ASR and also the extent of ASR degradation within a given area and control areas (non-wetted adjacent areas). Subject Matter Experts from around the country were consulted and a specific monitoring and action plan for ASR was added to the Structural Monitoring Program. Engineering evaluations that were performed and documented in foreign print 100716, (Subsection 3.8.6, Ref. 6) established that although the concrete can be considered degraded, the structures and embedded concrete anchors are capable of performing all required design basis functions. ASR is considered to be a degraded nonconforming condition pursuant to Regulatory Issue Summary (RIS) 2005-20. An operability determination was performed which established reasonable assurances that the structures and embedded/drilled-in concrete anchors are capable of performing all required design basis functions. Design basis calculations will be reconciled to account for ASR following completion of the actions delineated in the ASR corrective action plan.

3.8.4.7 Testing and In-Service Surveillance Requirements

Normal quality control testing is discussed in Subsection 3.8.3.6. A general visual inspection of the exposed accessible interior and exterior surfaces of the Containment Enclosure Building will be periodically conducted as discussed in Subsection 6.2.6.1.

The Structures Monitoring Program includes requirements for inspecting structures affected by ASR. The total expansion of ASR-affected areas is limited to less than the amounts specified in Table 3.8-18. Periodically verifying that ASR expansion levels are below the limits is necessary to ensure structural properties of ASR-affected areas are similar to areas with no evidence of ASR.

The Structures Monitoring Program also has limits on structure deformation from ASR concrete expansion. Structures with increasing levels of deformation, as determined by an analysis of the self-straining loads, are classified as Stage 1, 2, or 3. Monitoring criteria for each structure are included in the Structural Monitoring Program and inspection requirements are defined based on the analysis and classification of each structure.

3.8.4.7.1 Structures Monitoring Program

The Structures Monitoring Program is implemented through the plant Maintenance Rule Program, which is based on the guidance provided in NRC Regulatory Guide 1.160 "*Monitoring the Effectiveness of Maintenance at Nuclear power Plants*" and NUMARC 93-01 "*Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*", and with guidance from ACI 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures". The Structures Monitoring Program was developed using the guidance of these three documents. The Program is implemented to monitor the condition of structures and structural components within the scope of the Maintenance Rule, such that there is no loss of structure or structural component intended function.

3.8.4.7.2 Alkali-Silica Reaction (ASR) Aging Management

The ASR Aging Management Program manages cracking due to expansion and reaction with aggregates of concrete structures. The potential impact of ASR on the structural strength and anchorage capacity of concrete is a consequence of strains resulting from the expansive gel.

The Structures Monitoring Program performs visual inspections of the concrete structures at Seabrook for indications of the presence of alkali-silica reaction (ASR). ASR involves the formation of an alkali-silica gel which expands when it absorbs water. This expansion is volumetric in nature but is most readily detected by visual observation of cracking on the surface of the concrete. This cracking is the result of expansion that is occurring in the in-plane directions. Expansion is also occurring perpendicular (through the thickness of the wall) to the surface of the wall, but cracking will not be visible in this direction from the accessible surface.

Cracking on the surface of the concrete is typically accompanied by the presence of moisture and efflorescence. Concrete affected by expansive ASR is typically characterized by a network or "pattern" of cracks. Micro-cracking due to ASR is generated through forces applied by the expanding aggregate particles and/or swelling of the alkali-silica gel within and around the boundaries of reacting aggregate particles. The ASR gel may exude from the crack forming white secondary deposits at the concrete surface. The gel also often causes a dark discoloration of the cement paste surrounding the crack at the concrete surface. If "pattern" or "map" cracking typical of concrete affected by ASR is identified, an evaluation will be performed to determine further actions.

Monitoring of crack growth is used to assess the in-plane expansion associated with ASR and to specify monitoring intervals. In selected locations, cores will be removed for modulus testing to establish the level of through-thickness expansion to date. Instruments (extensometers) will be placed in the resulting bore holes to monitor expansion in this direction going forward.

ASR is primarily detected by non-intrusive visual observation of cracking on the surface of the concrete. The cracking is typically accompanied by the presence of moisture and efflorescence. ASR may also be detected or confirmed by removal of concrete cores and subsequent petrographic analysis.

A Combined Cracking Index (CCI) is established at thresholds at which structural evaluation is necessary (see table below). The Cracking Index (CI) is the summation of the crack widths on the horizontal or vertical sides of 20-inch by 30-inch grid on the ASR-affected concrete surface. The horizontal and vertical Cracking Indices are averaged to obtain a Combined Cracking Index (CCI) for each area of interest. A CCI of less than the 1.0 mm/m can be deemed acceptable with deficiencies (Tier 2). Deficiencies determined to be acceptable with further review are trended for evidence of further degradation. The change from qualitative monitoring to quantitative monitoring occurs when the Cracking Index (CI) of the pattern cracking equals or is greater than 0.5 mm/m in the vertical and horizontal directions. Concrete crack widths less than 0.05 mm cannot be accurately measured and reliably repeated with standard, visual inspection equipment. A CCI of 1.0 mm/m or greater requires structural evaluation (Tier 3). All locations meeting Tier 3 criteria will be monitored via CCI on a ½ year (6-month) inspection frequency and added to the through-thickness expansion monitoring via extensometers. All locations meeting the Tier 2 structures monitoring criteria will be monitored on a 2.5 year (30-month) frequency. CCI correlates well with strain in the in-plane directions and the ability to visually detect cracking in exposed surfaces making it an effective initial detection parameter.

Tier	Structural Monitoring Program Category	Recommendation for Individual Concrete Components	Criteria
3	Unacceptable (requires further evaluation)	• Structural Evaluation • Implement enhanced ASR monitoring, such as through-wall expansion monitoring using Extensometers.	1.0 mm/m or greater Combined Cracking Index (CCI)
2	Acceptable with Deficiencies	Quantitative Monitoring and Trending	• 0.5 mm/m or greater CCI • CI of greater than 0.5 mm/m in the vertical and horizontal directions.
		Qualitative Monitoring	Any area with visual presence of ASR (as defined in FHWA-HIF-12-022) accompanied by a CI of less than 0.5 mm/m in the vertical and horizontal directions.
1	Acceptable	Routine inspection as prescribed by the Structural Monitoring Program	Area has no indications of pattern cracking or water ingress- No visual symptoms of ASR

The Alkali-Silica Reaction Aging Management Program was initially based on published studies describing screening methods to determine when structural evaluations of ASR affected concrete are appropriate. Large scale destructive testing of concrete beams with accelerated ASR has confirmed that parameters being monitored are appropriate to manage the effects of ASR and that acceptance criterion of 1 mm/m a used provides sufficient margin.

CCI's limitation for heavily reinforced structures is that in-plane expansion, and therefore CCI, has been observed in the large scale test programs to plateau at a relatively low level of accumulated strain (approximately 1 mm/m). No structural impacts from ASR have been seen at these plateau levels in the large scale testing program at the University of Texas at Austin, Ferguson Structural Engineering Laboratory. While CCI remains useful for the detection and monitoring of ASR at the initial stages, an additional monitoring parameter in the out-of-plane direction is required to monitor more advanced ASR progression. ASR expansion in the out-of-plane direction will be monitored by borehole extensometers installed in drilled core bore holes. Through-thickness expansion will be monitored every six months, consistent with the monitoring interval for CCI of Tier 3 locations. This interval is appropriate because ASR is a relatively slow moving process and the concrete at Seabrook Station has been in existence for decades. There is no reason to expect a sudden acceleration of ASR development. Hence, structural implications will not change significantly in a short period of six months.

Although the observed strains due to ASR are of very small magnitude and adequately monitored by CCI and extensometers, over large distances and with the right building geometry, they can result in discernable dimension changes in a structure. Additional monitoring of this relative displacement potential and its impact to plant systems and components is included in the ASR Monitoring Program. Specifically, monitoring includes identifying signs of relative displacement or building deformation (e.g., fire seal displacement, seismic gap width changes, pipe/conduit misalignments at penetrations or between adjacent structures, bent or displaced pipe/conduit and supports, doorway misalignments). Critical building geometry locations where the potential for deformation is likely will be monitored for displacement via location-specific techniques.

3.8.4.7.3 Building Deformation Aging Management Program

The Building Deformation Aging Management Program is a plant specific program implemented under the existing Maintenance Rule Structures Monitoring Program. Building Deformation is an aging mechanism that may occur as a result of other aging effects of concrete. Building Deformation at Seabrook is primarily a result of the alkali silica reaction (ASR) but can also result from swelling, creep, and shrinkage. Building deformation can cause components within the structures to move such that their intended functions may be impacted.

The Building Deformation Aging Management Program uses visual inspections associated with the Structures Monitoring Program and cracking measurements associated with the Alkali-Silica Reaction program to identify buildings that are experiencing deformation. The first inspection is a baseline to identify areas that are exhibiting surface cracking. The surface cracking will be characterized and analytically documented. This inspection will also identify any local areas that are exhibiting deformation. The amount of components experiencing deformation and the extent of surface cracking will be input into an analytical model. This model will determine the extent of building deformation and the frequency of required visual inspections.

For building deformation, location-specific measurements (e.g. via laser target and gap measurements) will be compared against location-specific criteria to evaluate acceptability of the condition.

Structural evaluations will be performed on buildings and components affected by deformation as necessary to ensure that the structural function is maintained. Evaluations of structures will validate structural performance against the design basis, and may use results from the large-scale test programs, as appropriate.

Evaluations for structural deformation will also consider the impact to functionality of affected systems and components (e.g., conduit expansion joints). NextEra will evaluate the specific circumstances against the design basis of the affected system or component. Structural evaluations will be used to determine whether additional corrective actions (e.g., repairs) to the concrete or components are required. Specific criteria for selecting effective corrective actions will be evaluated on a location-specific basis.

3.8.5 Foundations

The following sections discuss the physical descriptions of the foundations, applicable codes, standards and specifications, loads and load combinations, design and analysis procedures, structural acceptance criteria, materials, quality control and special construction techniques, and testing and in-service inspection requirements for the foundations of seismic Category I structures.

3.8.5.1 Descriptions of the Foundations

The locations and relationships of the various seismic Category I foundations are shown on the plot plan, Figure 1.2-1. Details of individual foundations, including type and dimensions, are given in Table 3.8-15. This table also contains a list of figures showing plans and profiles of the foundations.

Foundations for seismic Category I buildings are conventionally reinforced concrete mats of varying thicknesses supported on sound bedrock or on fill concrete extending to sound bedrock. The walls of the Containment Enclosure Building extend to a spread footing, 10'-3" wide by 10' deep, which carries the load from the walls to sound rock. This footing is not continuous, having openings for the pipe chases and electrical tunnels below the Emergency Feedwater Pumphouse.

The bottoms of this footing and most mats are embedded in the rock in order to transfer horizontal shear forces. If the bottom of a foundation does not extend to sound rock, fill concrete was placed from the sound bedrock to the elevation of the underside of the structure.

The only exceptions to the above criteria are five safety-related electrical manholes, certain sections of service water pipes, and most of the 8350 feet of safety-related electrical duct banks which are supported on engineered backfill consisting of offsite borrow or tunnel cuttings (see Subsections 2.5.4.5c and 3.7(B).1.4). Compaction requirements for this engineered backfill ensure firm support which, along with flexible couplings between the manholes and ducts, prevent any seismic response in one of these components from inducing a response in an adjacent component or structure.

Since all other seismic Category I structures are founded on sound rock, or on fill concrete over sound rock, they are inherently isolated from each other. There is no connecting soil medium which could foster seismic coupling.

Similarly, the foundations of nonseismic Category I structures, which are also founded on rock, do not have seismic coupling with the foundations of seismic Category I structures, except for the Circulating Water Pumphouse. This is a non-Category I structure which is attached to the Service Water Pumphouse, shares a common foundation it, and it is designed as a unit. Seismic isolation of the buildings above the point of fixity is ensured by means of isolation joints; these joints are provided between adjacent buildings above the point of fixity.

The point of fixity is the elevation on the building to which rock extends or fill concrete was placed. There is no correlation between this point and the natural rock elevation; it may be lower than natural rock or higher if fill concrete was used to increase the embedment. Within the region of a building that is fixed, the walls and foundation are rigid and accelerate along with the rock.

There are no unique features used for foundations.

The load transfer for the foundations is as follows:

a. Load is applied to the wall of a seismic Category I structure.

b. The wall transfers the load to the foundation (mat or footing).

c. The foundation transfers the load to the rock, fill concrete, or engineered fill.

A more detailed description of the mechanism of load transfer is given in Subsection 3.8.5.4.

For a detail of the typical reinforcing pattern at the junction of a reinforced concrete wall and its foundation, see Figure 3.8-36.

Piles are not used under seismic Category I structures. The structures transmit load directly to rock by contact or through fill concrete or engineered fill.

3.8.5.2 Applicable Codes, Standards and Specifications

For applicable codes, standards and specifications, see Subsections 3.8.1.2 and 3.8.4.2 for the containment structure foundation and for other seismic Category I structure foundations, respectively.

3.8.5.3 Loads and Load Combinations

For the containment structure foundation, see Subsection 3.8.1.3 for loads, load combinations, load factors and the design approach used with the load combinations and load factors.

For other seismic Category I structure foundations, see Subsection 3.8.4.3 for loads, load combinations, load factors and the design approach used with the load combinations and load factors.

Foundations and structures are checked for sliding and overturning due to earthquakes, winds, and tornadoes and for flotation due to floods and high water table using load combinations described in Subsections 3.8.1.3 and 3.8.4.3 for the containment structure and the other seismic Category I structures, respectively.

Lateral earth pressures are considered as applicable. Nonrigid walls are designed for active earth pressure under static conditions and for an equivalent static earth pressure based on a coefficient of dynamic earth pressure that dampens with depth under seismic conditions. Rigid walls are designed for earth pressure at rest under static conditions and for an equivalent static earth pressure based on a constant coefficient of dynamic earth pressure under seismic conditions. Lateral pressure diagrams for these conditions are shown in Figure 2.5-52 and Figure 2.5-53 and discussed in Subsection 2.5.4.11.

3.8.5.4 Design d Analysis Procedures

The foundations of seismic Category I structures are analyzed and designed in accordance with Subsections 5.2 and 3.8.5.5 to determine maximum stresses in reinforcing and concrete, using the load combinations discussed in Subsection 3.8.5.3.

a. Boundary Conditions and Expected Behavior

Most foundations of seismic Category I structures are founded directly on sound rock or on fill concrete; five safety-related electrical manholes and most of the safety-related electrical duct banks are supported on engineered fill. The entire length of the duct banks, however, is designed with the assumption of support on engineered fill.

Design and analysis, including idealization and boundary conditions, for the circular base mat of the containment structure are described in Subsection 3.8.1.4. The base mat is designed to sustain all credible loads resulting from the containment and internal structures.

Design procedures for all structures insured that foundation mats and footings were sized to limit bearing pressure on the rock to 60 tons/square foot on horizontal surfaces and 10 tons/square foot in weathered zone and 60 tons/square foot below weathered zone on vertical surfaces. These bearing pressures were established on the basis of results of tests of unconfined compressive strength with a factor of safety on the order of 10. Foundations for those structures supported on engineered fill are designed to limit bearing pressures to the allowable limits given in Subsection 2.5.4.5.

Each building has an individual foundation; no common foundations are employed except for the Service and Circulating Water Pumphouse. In a few additional cases, as noted in Subsection 3.8.4.1, a structure is partitioned into a dual-function structure which is designed as a unit. Also, adjacent foundations such as those for the containment structure and the Enclosure Building, even though they are not continuous, may transfer horizontal forces through bearing.

All loads are transferred to the foundation through the walls, or in the cases of some equipment, through the supporting pedestals. Reinforcing continues from the wall into the foundation mat and is embedded in the mat and wall by hooks, bends or straight extensions of sufficient length to develop the design strength of the bars. Thus, forces are adequately transferred from the wall to the foundation.

Soil-structure interaction is not applicable for rock-supported seismic Category I structures. Thus, for these structures, the foundation mats are designed as flat plates supported on rigid non-yielding foundation media.

In the case of electrical duct banks in engineered fill, the ability of the elastic fill to impose a seismically induced strain on the duct bank was considered. This is further described in Subsection 3.7(B).2.

For safety-related electrical manholes which are supported on engineered fill, amplification of the ground motion is considered. The slab, however, is essentially rigid, and bearing pressures are very low. (See Subsection 3.7(B).2.)

Behavior of the foundation mats when subjected to the load combinations given in Subsection 3.8.5.3 is within the acceptance criteria of Subsection 3.8.5.5.

b. <u>Vertical Loads, Lateral Loads and Overturning Moments</u>

Vertical loads are carried by direct bearing on the rock. Overturning moments and bearing pressures on the rock (including maximum toe pressure) were investigated in accordance with the safety factors discussed in Subsections 3.8.1.3 and 3.8.4.3 for the containment foundations and other seismic Category I structure foundations, respectively.

Base shears and torsional moments are transferred to rock by one or both of the following:

1. Embedment in rock which provides resistance to shearing loads through direct bearing

2. Friction between the bottom of the foundation and the top of the rock (the coefficient of friction takes into consideration the reduced shear resistance due to the presence of the waterproofing membrane).

In the containment structure, the walls of the reactor cavity bear against the rock through fill concrete to provide additional resistance for transfer of base shears. The walls of the cavity are designed to carry these loads.

Conventional hand calculations were employed for the design of the foundations and were augmented, for some of the larger foundations, by finite element techniques to determine uplift patterns.

This uplift was determined by static methods using forces which were determined in the dynamic seismic analysis. The mat was modeled by a series of flat plate bending elements which include stiffening effects of the walls or shell. An uplift pattern was assumed, loads applied, and final displacements checked. When the calculated displacements were in agreement with the assumed uplift pattern, the analysis for that load combination was then completed by calculating shears and bending moments.

Methods of determining the overturning moments which were used in the uplift analysis include all three components of the earthquake and are discussed in Subsection 3.7(B).2.

Design factors of safety against sliding, overturning and buoyancy are defined in Subsections 3.8.1.3 and 3.8.4.3 for the containment structure and other seismic Category I structures, respectively.

c. Computer Programs

See Appendix 3F for a description of the programs. See Table 3.8-17 for a list of structures and the computer programs which were used for them.

3.8.5.5 Structural Acceptance Criteria

The acceptance criteria relating to stress, strain, gross deformation and shear loads are described in Subsections 3.8.1.5 and 3.8.4.5 for the containment and other seismic Category I structure foundations, respectively.

Safety factors for buoyancy, sliding, and overturning are as follows:

	Factor of Safety		
Load	Overturning	Sliding	Flotation
Service/Normal load combinations	1.5	1.5	--
Factored/Unusual load combinations	1.1	1.1	--
Dead load and design basis flood load	--	--	1.5

3.8.5.6 Materials, Quality Control and Special Construction Techniques

The primary materials of construction are concrete and reinforcing steel. Their descriptions and basic quality control procedures are discussed in Subsections 3.8.1.6 and 3.8.4.6 for the containment foundation and other seismic Category I structure foundations, respectively. Engineered fill, fill concrete and backfill concrete are described in Subsection 2.5.4.5.

There are no special construction techniques.

3.8.5.7 Testing and In-Service Surveillance Requirements

The ability of the containment foundation to resist 1.15 times the design pressure is demonstrated during the structural integrity test as described in Subsection 3.8.1.7.

For other seismic Category I structure foundations, no preoperational or in-service surveillance is required.

Structures which are founded on sound rock or on fill concrete over sound rock do not have any potential areas of settlement or displacement which should be monitored. Similarly, gradation requirements, compaction criteria and compaction tests for engineered fill ensure a foundation material which will support the design loads with negligible settlement. Piles were not used. For these reasons there are no potential settlements or displacements which should be monitored for any foundation.

3.8.6 References

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7. MPR-4288, Revision 0, "Seabrook Station: Impact of Alkali-Silicia Reaction on the Structural Design Evaluations," July 2016. FP#101020

8. MPR-4273, Revision 1, "Seabrook Station – Implications of Large-Scale Test Program Results on Reinforced Concrete Affected by Alkali-Silica Reaction." FP#101050

9. MPR-4153, Revision 3, "Seabrook Station – Approach for Determining Through-Thickness Expansion from Alkali-Silica Reaction." FP#100918

10. SGH Document 170444-MD-01, Revision 1, "Methodology for the Analysis of Seismic Category I Structures with Concrete Affected By Alkali-Silica Reaction." FP#101196

3.9(B) MECHANICAL SYSTEMS AND COMPONENTS

3.9(B).1 Special Topics for Mechanical Components

3.9(B).1.1 Design Transients

Refer to Subsection 3.9(N).1.1.

3.9(B).1.2 Computer Programs Used in Analyses

Computer programs used in dynamic and static analyses of seismic Category I code and noncode items are described below for their theories, assumptions, references and applications.

a. ADLPIPE

The stresses and loads in piping systems due to thermal expansion, dead weight, and seismic loadings are calculated by using the ADLPIPE program.

ADLPIPE is a computer program developed by Arthur D. Little, Inc., of Cambridge, Massachusetts, and is used for static and dynamic elastic analyses of complex piping systems. The piping is modeled as a series of sections that lie between network points. A section is composed of a sequence of straight and/or curved pipes, and each pipe may have common or different loads and physical properties. The network points may be free, partially or fully restrained, and have specified displacements that represent thermal anchor displacements or seismic anchor motion. Intermediate springs to ground or joining other members may be placed within the section to represent spring hangers, pipe bellows, skew and guided restraints, supports and equipment. The matrix displacement method is used throughout the program, and transfer matrix techniques are used to reduce the size of the system stiffness matrix.

The ADLPIPE computer program provides the ability to compute a summary stress report which can be used for direct comparison with the criteria of ASME Section III.

Verification test cases for ADLPIPE have been performed by UE&C's computer service vendor. Test problems have been executed and documented which agree with the program author's verification documents. Acceptable verification criteria are: (1) hand calculations, (2) known theoretical and experimental solutions and (3) solutions from other verified computer programs.

b. MRI/STARDYNE

This computer program uses the finite-element method for the static and dynamic analysis of two- and three-dimensional linear elastic structural models, subjected to any arbitrary static or dynamic loading. Static results and dynamic responses can be presented as structural deformations, and/or internal member loads/stresses. Using either the direct integration or the normal mode techniques, dynamic response analysis can be performed for a wide range of loading conditions, including transient, steady-state harmonic, random and shock spectra excitation.

The MRI/STARDYNE code is used in the analysis of seismic Category I cable tray and conduit systems.

Verification for the STARDYNE program was accomplished in the same manner as that used for ADLPIPE.

c. ANSYS

ANSYS is a large-scale general purpose computer program. The matrix displacement method of analysis based upon finite element idealization is used throughout the program. Analytical capabilities include static and dynamic, small and large deformations, steady-state and transient heat transfer and steady-state fluid flow. Loading may be forces, displacements, pressures and temperatures.

Verification for the ANSYS program was accomplished in the same manner as that used for ADLPIPE.

d. ICES/STRUDL-II

This computer program provides the ability to perform static and dynamic analysis for framed structures and three-dimensional solid structures. The matrix displacement method of analysis based on finite element idealization is used throughout the program.

Verification for the ICES/STRUDL-II program was accomplished in the same manner as that used for ADLPIPE.

e. ADLPIPE-2

A static and dynamic pipe design and stress analysis program developed by Arthur D. Little, Inc., and modified by UE&C, the program provides elastic analyses of piping systems in accordance with the requirements of ASME Section III - Nuclear Power Plant Components, ANSI B31.1 - Power Piping and USNRC Regulatory Guide 1.92.

Analytical capabilities of the program include:

- Deadweight
- Thermal
- Anchor movements
- Pressure
- External forces
- Seismic response spectra including closely spaced modes (Regulatory Guide 1.92).

The validity of ADLPIPE-2 results has been verified by comparing results of identical problems analyzed by other verified computer codes or hand calculations. Computations of forces, moments and displacements have been verified against the original ADLPIPE program, the validity of which is supported by Arthur D. Little, Inc., in documentation of ADLPIPE. The ADLPIPE-2 program was further verified in 1979 by a series of benchmark piping problems presented by the NRC in BNL-NUREG-21241-R2. Supplemental verification has been achieved by demonstrating exceptionally good agreement between ADLPIPE-2 results and the results published for Problem #4 in NUREG/CR-1677. Comparisons of elemental forces, reaction forces, moments, displacements, eigenvalues, eigenvectors, and stress computations were made. Computations for determining the results of a three component earthquake using square-root-of-the-sum-of-the-squares (SRSS) summations, closely spaced modes and stress intensification factors have been verified by hand calculations. All of the above verification efforts have been fully documented and are available for review.

This program has been used for analysis of ASME Section III, Class 2 and 3, and ANSI B31.1 piping systems.

f. IMAPS

The IMAPS (Interactive Mode Analysis of Piping and Structures) computer code performs static linear elastic analyses of three-dimensional multisupported simplified piping systems of framed structures.

Analytical capabilities of the program include:

- Deadweight
- Thermal
- Anchor movements
- External forces and moments
- Seismic static.

The validity of the results for IMAPS has been verified by comparison of results for identical problems performed on the ADLPIPE-2 program. In addition, IMAPS results have been compared with benchmark piping problems from USNRC (BNL-NUREG-21241-R2) with excellent comparison of results.

This program has been used for analyses of small ASME Section III, Class 2 and 3, and ANSI B31.1 piping.

g. SPHNOZ/CYLNOZ

This program computes local stresses in spherical and cylindrical shells due to external loads. The program is based on the graphical results of the analytical work by Prof. Bijlaard, published in Welding Research Council Bulletin No. 107, Rev. 2, July 1970, Franklin Institute Research Laboratories, Philadelphia, Pa.

The program results have been verified by comparison with hand calculations based on WRC Bulletin No. 107.

This program has been used as an alternative method to certify the design adequacy of branch piping and nozzles.

h. WATER

Program WATER determines the transient forces which result from rapid opening of a safety/relief valve in a piping system. Initially, water is upstream from the valve and air is downstream from the valve. Water discharges through the valve, and downstream conditions are such that flashing does not occur. Program output consists of total force vs. time history at each elbow in the downstream piping system, or, equivalently, the net force/time history on each piping leg.

The program is based upon in-house formulation using rigid water column theory. Verification consists of demonstrating that program results and those obtained from a hand calculation agree for identical input data. Additionally, fluid output parameters (acceleration and velocity) from program WATER agrees with results from program RELAP. Program RELAP is in the public domain.

i. VALCLO

Program VALCLO determines the maximum force developed due to closing a valve with prescribed closing characteristics. Prior to valve closing, steam is flowing in the system. The maximum force is determined based upon formulas presented in "Steam Hammer in Turbine Piping Systems," Coccio, C.L.; ASME Publication 66-WA/FE-32, 1968.

Verification consists of demonstrating that computer results and results obtained from a hand calculation agree for identical input data.

j. ELBFOR

Program ELBFOR is a wave super-position program. It is run as a post processor to program VALCLO; however, ELBFOR could be used as a post processor to any source calculation which would establish the maximum force in the system. Using the maximum force, ELBFOR establishes a ramp forcing function over the valve operating time. Considering system geometry, the ramp forcing function is propagated through the system at sonic velocity to establish the total force/time history for each elbow in the system, or, equivalently, the net force/time history for each piping leg in the system.

Verification of ELBFOR consists of demonstrating that computer results and results obtained from a hand calculation agree for identical input data.

k. MADIS

Program MADIS determines the transient forces which result from rapid opening of a safety/relief valve in a piping system. Initially, superheated or saturated steam is upstream from the valve and air is downstream from the valve. The program output consists of the total force/time history at each elbow in the downstream piping system or, equivalently, the net force/time history on each piping leg. The program is based upon formulas presented in "Analysis of Safety Valve Discharging into Closed Piping System;" Luk, C.H.; Structural Design of Nuclear Plant Facilities, No. 1; Specialty Conference held at Chicago, Ill. on December 17-18, 1973.

Verification of the program consists of demonstrating that computer results and those obtained from hand calculations agree for identical input data.

l. FLEXPLT

FLEXPLT is an interactive time-sharing computer program used for the analysis of flexible baseplates. These baseplates have concrete expansion anchors as the means of attachment to the concrete. A variety of static load combinations may be applied through a single structural attachment.

FLEXPLT has the capability of analyzing the baseplate with the structural attachment centrally located on the plate and the anchors within a given window, or it can analyze the baseplate with the anchors at any fixed location and the structural attachment off the centerline of the baseplate.

FLEXPLT has been verified by comparison of results for identical problems using the PSBASEPLATE computer program and hand calculations.

m. GT STRUDL

GT STRUDL is a large-scale general purpose computer program. The matrix displacement method of analysis based upon finite element idealization is used throughout the program. GT STRUDL has the ability to perform static and dynamic analysis for framed structures and three-dimensional solid structures.

GT STRUDL is used in the analysis and design of nuclear and nonnuclear linear type pipe supports and seismic Category I duct supports.

The validity of results for GT STRUDL has been verified by comparison of results for identical problems performed using hand calculations.

n. PSBASEPLATE

PSBASEPLATE is a pre- and postprocessing computer program which interfaces with STARDYNE. PSBASEPLATE is used for analyzing flexible baseplates on a geometrically nonlinear foundation. PSBASEPLATE can generate the finite element model of a single structural attachment to a baseplate fastened to ground by concrete expansion anchor.

PSBASEPLATE processes the output produced by STARDYNE to aid the user in interpreting the results.

The validity of results for PSBASEPLATE has been verified by comparison of results for identical problems using other recognized computer codes.

o. BASEPLATE II

BASEPLATE II is a pre- and postprocessing computer program which interfaces with STARDYNE. BASEPLATE II is used for analyzing flexible baseplates on a geometrically nonlinear foundation. BASEPLATE II can generate the finite element model of either the plate-bolt-structure assembly, or a model including a framed structure attached to the baseplate.

BASEPLATE II processes the output produced by STARDYNE to aid the user in interpreting the results.

The validity of results for BASEPLATE II has been verified by comparison of results for identical problems using other recognized computer codes.

p. McAuto STRUDL

McAuto STRUDL is a large-scale general purpose computer program. The matrix displacement method of analysis based upon finite element idealization is used throughout the program. McAuto STRUDL has the ability to perform static and dynamic analysis for framed structures and three-dimensional solid structures.

McAuto STRUDL has extensive postprocessing capabilities, including an ASME Section III, Subsection NF code check for linear type supports. Also, McAuto STRUDL has the capability to design a limited class of fillet welds in butt joint connections.

McAuto STRUDL is used in the analysis and design of nuclear and nonnuclear linear type pipe supports.

The validity of results for McAuto STRUDL has been verified by comparison of results for identical problems performed using hand calculations or other recognized computer codes such as NASTRAN and ANSYS.

q. STFPSG

STFPSG is an interactive, time-sharing computer program using closed form solutions to calculate the structural stiffness or moment of inertia of a limited class of structural frames and beams.

STFPSG has been verified by comparing the computer-generated values with hand-calculated values.

r. PSGPREP

PSGPREP is an interactive, time-sharing computer program, which will prepare input data for a McAuto STRUDL analysis from user keyboard input.

PSGPREP is designed to produce input suitable for an analysis of a pipe support. However, PSGPREP can be used to handle analyses of other structural frames.

PSGPREP has been verified by comparing its results to manually generated input.

s. PSGWARP

PSGWARP is an interactive, time-sharing computer program which calculates the stresses in beams due to torsional effects. The closed form solutions employed in PSGWARP were extracted from the Bethlehem Steel publication "Torsion Analysis of Rolled Steel Sections."

PSGWARP will also, by superposition, combine bending and shear stresses with torsional stresses and compare them with ASME Code allowables (Subsection NF).

PSGWARP was verified by comparing results with those from identical input supplied to the McAuto STRUDL Program.

t. CHADS

CHADS is an interactive, time-sharing program that enables a user to create a pipe support mathematical model including loading conditions, member properties, material properties, and code checking data to generate input for a GTSTRUDL analysis. The user can also employ the same geometry data to create a complete detail drawing of the support and then use GTSTRUDL and WELDDA to check member stress and weld design.

CHADS also uses the GTSTRUDL output to prepare input for BASEPLATE II, if it is desired to analyze an anchorage consisting of concrete expansion bolts and plates.

Verification of CHADS was accomplished in the same manner as for ADLPIPE.

u. WELDDA

WELDDA is a postprocessing program to the GTSTRUDL computer code for designing and analyzing welded connections between rolled structural members. WELDDA uses a closed form solution based on analyzing the weld pattern as a line to calculate weld stress and/or size. WELDDA then performs a code check to either the AISC or ASME Codes.

Verification of WELDDA was accomplished in the same manner as for ADLPIPE.

v. DIS (Design Information System)

DIS is an interactive piping data management system developed by Arthur D. Little, Inc., for piping engineering. Equipment, pipe, and fabrication specifications are stored in a data base. From these specifications, a piping analytical isometric can be developed. DIS can also be used to generate a bill of material and the ADLPIPE mathematical model. The user also controls printing of various output reports.

DIS was verified in the same manner as ADLPIPE.

w. NUPIPE II

The stresses and loads in piping systems due to thermal expansion, dead weight, and seismic loadings are calculated by using the NUPIPE II program.

NUPIPE II is a computer program developed by Quadrex Corporation of Campbell, California, and is used for static and dynamic elastic analyses of complex piping systems. The matrix displacement method of analysis based upon finite element idealization is used throughout the program. The piping is modeled as individual elements connected by node points. The node points may be free, fully or partially restrained, have specified displacements, or, connected to ground through an element with a finite stiffness to represent pipe supports or equipment. Spring stiffnesses may be used to connect elements to represent pipe bellows or ball joints.

Verification test cases for NUPIPE II have been performed by UE&C's computer service vendor. Test problems have been executed and documented and agree with the program author's verification documents. Acceptable verification criteria are: (1) hand calculations, (2) known theoretical and experimental solutions and (3) solutions from other verified computer programs.

Verification for NUPIPE II was accomplished in the same manner as that used for ADLPIPE.

x. CONCYSE

CONCYSE is an interactive, structural, finite element program for nuclear power plant applications. The program is used in the analysis of structural members, baseplates, and welds. CONCYSE has extensive plotting capabilities and can combine output files into a complete analysis report for the structure.

CONCYSE allows the finite element model to be created and inspected interactively which reduces the possibility of analyzing an incorrect model. Added features are an ASME, Subsection NF Code Check on the structural members and an interactive program to analyze or design the welded connections of the structure.

Verification of CONCYSE was performed by UE&C's computer service vendor and is consistent with UE&C General Administrative Procedure 19 (GAP-0019), Rev. 2, September 16, 1985.

3.9(B).1.3 Experimental Stress Analysis

The experimental stress analysis method is not used for seismic Category I ASME Code and non-Code mechanical components. The analytical method is used instead.

3.9(B).1.4 Considerations for Evaluation of the Faulted Condition

a. Seismic Category I Components

All seismic Category I mechanical components were evaluated for the faulted loading conditions. In most cases, conservative stress limits such as the stress limits for the upset plant condition were applied to the faulted condition. Actual stresses are within elastic limits. The inelastic method of analysis was not used to evaluate the design of safety-related Code or non-Code items for the faulted condition. For Code or non-Code mechanical equipment supports, elastic analysis methods were used for evaluating faulted loading conditions, and the stress limits for mechanical supports are within the elastic range of the material used in the fabrication of the support. The rules employed for evaluation of faulted condition loading effects for supports of ASME Code mechanical equipment are in accordance with either Subsection NF of the ASME B&PV Code, Section III, or AISC criteria which provide comparable stress criteria.

b. Piping

1. Faulted Condition Under Pressure-Retaining Integrity

All seismic Category I piping systems were evaluated for the faulted loading condition to provide assurance of pressure-retaining integrity. The elastic method of analysis was used according to the rules of the ASME Code, Section III, Appendix F. The effect of the anchor displacements and temperature conditions for the faulted condition on the piping systems was evaluated, using a static method of analysis for displacement and thermal expansion. Specifically, the following acceptance criteria were used for the various piping systems during the faulted condition:

(a) Class 1 piping systems were designed and analyzed for the load combinations and stress limits of Table 3.9(B)-8.

(b) Class 2 and 3 essential piping systems which are required to retain functional integrity were designed and analyzed for the load combinations and stress limits of Table 3.9(B)-9. The check of equations (9), (10) and (11) of NC/ND-3652 assured that piping deformation due to anchor displacements and thermal effect of fluid or environmental temperature during the faulted condition will not impair the functional integrity of the systems.

2. <u>Faulted Condition under Structural Integrity</u>

Class 2 and 3, and nonclass, nonessential piping systems which are required to retain structural integrity were designed and analyzed for the load combinations and stress limits of Table 3.9(B)-10. The check of primary stress limits assured that structural integrity of the piping will not be impaired. The effect of environmental temperature on the piping system was evaluated in terms of additional loads on supports and anchors, due to expansion of piping predicted on the basis of elastic properties of the piping. The expansion stress limits were not checked for piping in this category since structural deformation induced by self-limiting load is permitted (expansion stresses which result from the constraint of free-end displacement).

3. <u>Faulted Condition under Pipe Break</u>

Refer to Subsections 3.6(B).2 and 3.6(N).2 for a discussion on this subject.

c. <u>Supports</u>

For evaluation of supports under the faulted condition, refer to Subsection 3.9(B).3.4.

3.9(B).2 Dynamic Testing and Analysis

3.9(B).2.1 <u>Piping Vibration, Thermal Expansion and Dynamic Effects</u>

All ASME Code Class 1, 2 and 3 piping for Seabrook Station has been designed, analyzed, fabricated and erected in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code. Design and analysis of the piping systems included consideration of thermal, deadweight, pressure, seismic and operating transient loads.

The following tests will be performed on ASME Class 1, 2 and 3 systems, high-energy systems located inside seismic Category I structures, high-energy systems whose failure could reduce the function of any seismic Category I feature to an unacceptable level, and seismic Category I portions of moderate energy systems located outside containment.

a. <u>Vibration</u>

As part of the preoperational testing program, transient and steady-state vibration of high-energy critical piping will be monitored and evaluated. The following testing program will be implemented to test and evaluate the vibration effect:

1. A list of the systems, flow modes of operation and transients to which each system will be tested is contained in Table 3.9(B)-1.

2. Small bore piping and instrumentation lines connected to the main piping will be included in the testing program.

3. For all system lines within the scope of the systems identified in Table 3.9(B)-1, the vibration levels will be analyzed. At selected locations, the peak-to-peak displacements will be measured. Inaccessible locations and noninstrumented transients will be visually observed.

4. If measured parameters exceed the acceptance criteria, the effect of the vibration on design will be determined by further evaluation. Areas of evaluation are as follows:

   (a) Fluid system parameters, pressure, flow rate and temperature

   (b) Frequency of vibration.

   Following further evaluation, corrective action will be initiated until the system meets the acceptance criteria.

5. For steady-state vibration, the piping break stress (zero to peak) due to vibration only (neglecting pressure) will not exceed 12,000 psi for austenitic stainless steel and 7,690 psi for carbon and low alloy steels. These limits are below the piping material fatigue endurance limits as defined in the design fatigue curves in Appendix I of ASME code for 10^6 cycles.

b. <u>Thermal Expansion</u>

During hot functional testing, the temperature of the reactor coolant and main steam piping will be raised to the operating temperature in increments of 100°F.

At selected temperatures, the thermal expansion of the piping will be checked at predetermined points. If thermal motion is not as predicted, the support system will be examined to verify correct function or to locate points of binding of restraints. If binding is found, the restraints will be adjusted to eliminate the unacceptable condition or reanalyzed to verify that the existing condition is acceptable.

If the support system is found to be functioning properly, but thermal expansion measurements vary from predicted values by more than a reasonable tolerance, then the analysis will be reviewed to explain the anomaly.

Reasonable tolerances, for the purpose of these tests, are as follows:

1. For expected motions of 1/8 inch or less, no measurements are anticipated. In this case a tolerance of ± 1/8 inch is acceptable.

2. For expected motions larger than 1/8 inch but less than 2 inches, a tolerance of ±50 percent is acceptable.

3. For expected motions greater than 2 inches, a tolerance of ±25 percent of the expected motion is acceptable.

Systems and locations requiring monitoring of thermal expansion during startup functional testing are listed in Table 3.9(B)-2.

c. Dynamic Effects

During the preoperational test and initial startup programs, the pumps will be started and stopped and valves will be stroked to verify proper operation. The effects of the transient vibrations and shocks resulting from these operations will be visually observed to verify that no excessive motion of piping or equipment results. If excessive motions are observed as a result of these operations, the support systems will be adjusted to eliminate the unacceptable motion. In cases where sequence of operations or operating procedures are found to cause unacceptable conditions, procedures will be changed to eliminate the unacceptable conditions.

For operating transient vibration, the piping bending stress, (zero to peak) due to the operating transient vibration, will be correlated with the calculated stress level from the piping stress report to ensure that the calculated total primary stress level is maintained within acceptable code levels. Where testing results indicate unacceptable levels, action will be taken to reduce or eliminate the effect of the operating transient vibration.

The details of the piping vibration, thermal expansion and dynamic effects testing will be incorporated into the individual test procedures for conducting these tests, as abstracted in Chapter 14. Specific information and data will be generated for the preparation of piping tests. This will include the different flow modes, identification of the selected locations for visual inspections and measurements, the acceptance criteria, and the possible corrective actions if excessive vibration occurs.

3.9(B).2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

Seismic Category I safety-related mechanical equipment has been designed to withstand the combined normal operating accident loads concurrent with the effects of the Safe Shutdown Earthquake (SSE) or the Operating Basis Earthquake (OBE). Verification of the adequacy of the equipment to withstand these combined loads has been demonstrated by analysis, by testing, or a combination of analysis and testing. Selection of the method of qualification was determined by the type, size, shape and complexity of the equipment being considered.

Operability of active Class 1, 2 or 3 valves, active Class 2 and 3 pumps, and their vital auxiliary equipment has been demonstrated by the combination of analysis and testing. Results of the qualification tests are given in Subsection 3.9(B).3.2. The seismic qualification procedures used for the operability verification of other active mechanical equipment are similar to the procedures described in Subsection 3.9(B).3.2 for pumps and valves.

For the mechanical equipment which is mechanically or structurally too complex, i.e., the seismic response cannot be adequately predicted analytically, testing procedures similar to those described in Section 3.10 were used to demonstrate the equipment operability and adequacy.

The structural and functional integrity of nonactive seismic Category I mechanical equipment has been demonstrated by one of the following two methods:

a. By analytical methods, to satisfy the stress criteria applicable to that specific piece of equipment

b. By seismic testing, to show that the equipment retains its structural and functional integrity under the simulated test environment

   This method is not applicable to pressure retaining boundaries of equipment since the stresses at the pressure boundary must be evaluated by analysis, as required by Section III of the ASME Boiler and Pressure Vessel Code.

The analytical or testing qualification methods discussed above are similar to the procedure described in Subsection 3.9(B).3.2, except that no functional verification is required.

Generally, the test procedures are based upon IEEE STD 344-1975 using random input motion. For the purely mechanical equipment, test procedures for a limited number of components were based upon IEEE STD 344-1971, using sinusoidal input motion and including a 1.5 factor applied to the motion. This results in compliance with IEEE STD 344-1975.

3.9(B).2.3 Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions

Refer to Subsection 3.9(N).2.3.

3.9(B).2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Refer to Subsection 3.9(N).2.4.

3.9(B).2.5 Dynamic System Analysis of the Reactor Internals under Faulted Conditions

Refer to Subsection 3.9(N).2.5.

3.9(B).2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

Refer to Subsection 3.9(N).2.6.

3.9(B).3 ASME Code Class 1, 2 and 3 Components, Component Supports and Core Support Structures

3.9(B).3.1 Loading Combinations, Design Transients, and Stress Limits

The load combinations and the design stress limits associated with the plant operating conditions which are applied to the design and analysis of the ASME III Code-constructed items, other than the NSSS items, are defined herein. The plant conditions considered were normal operation, postulated accidents and specified seismic events. Design transients are further discussed in Subsection 3.9(B).1.1. The requirements of ANSI/ANS-51.1-1983 have been satisfied by use of plant conditions and allowable stress limits imposed on active and nonactive components.

For the non-ASME component members (other than bolts) of the ASME III Code-constructed items, the design criteria limit the principal stresses to 0.6 Fy for the plant upset conditions, and to 0.9 Fy for the plant faulted condition. Refer to Subsection 3.9(B).3.4c for the bolt design criteria.

a. Valves, Pumps, Heat Exchangers, and Tanks

The loading conditions considered (where applicable) for the design of ASME Class 1, 2 and 3 components included, but were not limited to, loading effects resulting from:

1. Internal and external pressure

2. Dead load, i.e., weight of the component and normal contents, including additional pressure due to static and dynamic head of liquid

3. Superimposed loads caused by other components, such as nozzle loads

4. Environmental loads, wind loads, snow loads, and seismic loads for both an OBE and a SSE

5. Valve thrust and moments

6. Thermal and thermal transients (for Class 1 components only).

The loading combinations considered (where applicable) in the design and analysis of the ASME Code Class 1, 2 and 3 and certain non-Code safety-related components were categorized with respect to plant operating conditions defined as normal, upset, emergency and faulted conditions, as identified in Table 3.9(B)-3. The corresponding stress limits for each category of plant operating condition are presented in Table 3.9(B)-4 for nonactive pumps, Table 3.9(B)-5 for nonactive valves, Table 3.9(B)-6 for nonactive Class 1 valves and Table 3.9(B)-7 for ASME Code Class 2 and 3 pressure vessels and storage tanks. The stress limits for active pumps and valves are discussed in Subsection 3.9(B).3.2. The stress limits established for the various components are sufficiently low so that violation of the pressure boundary will not occur.

b. Piping Systems, Including In-Line Valves

The safety-related piping systems have been designed to satisfy the appropriate stress limits of the ASME III Code and those of Regulatory Guide 1.48, as delineated below:

1. For those piping systems that constitute a portion of the reactor coolant pressure boundary and have been designated as ASME III, Class 1 lines, the load combinations and stress limits for various plant operating conditions are presented in Table 3.9(B)-8.

   The following are the ASME Code Class 1 pipes qualified by the original A-E, UE&C (see Subsection 3.9(N).1.1 for design transient list applicable to Class 1 components):

Line No.	Line Size	Line Description	P&ID
91-1	1"	Reactor vessel vent line	RC-20845
91-2	1"	Reactor vessel vent line	RC-20845
328-6	2"	From seal inject. Filters to RC-P-1A	CS-20726
328-7	1½"	From seal inject. Filters to RC-P-1A	CS-20726
329-4	2"	From seal inject. filters to RC-P-1B	CS-20726
329-5	1½"	From seal inject. filters to RC-P-1B	CS-20726
330-4	2"	From seal inject. filters to RC-P-1C	CS-20726
330-5	1½"	From seal inject. filters to RC-P-1C	CS-20726
331-4	2"	From seal inject.filters to RC-P-1D	CS-20726
331-5	1½"	From seal inject. filters to RC-P-1D	CS-20726
80-1	6"	Pressurizer vent line	RC-20846
80-2	3"	Pressurizer vent line	RC-20846

Line No.	Line Size	Line Description	P&ID
80-15	6"	Pressurizer vent line	RC-20846
80-6	3"	Pressurizer vent line	RC-20846
74-1	6"	Suction line of pressurizer relief valve	RC-20846
75-1	6"	Suction line of pressurizer relief valve	RC-20846
76-1	6"	Suction line of pressurizer relief valve	RC-20846

Westinghouse has responsibility for Class 1 component core support structures and specific Class 1 piping. UE&C has responsibility for pressurizer safety relief line, the reactor coolant system drain line and Class 1 reactor coolant pump seal piping.

2. For those essential piping systems which have been designated as ASME III, Class 2 and 3, and which are required for safe shutdown of the reactor, the load combinations and stress limits for various plant operating conditions are presented in Table 3.9(B)-9.

3. For those nonessential piping systems which have been designated as ASME III Class 2 and 3, but which are not required for safe shutdown of the reactor, the load combinations and stress limits for various plant operating conditions are presented in Table 3.9(B)-10.

   Definitions of the symbols and notations used in Table 3.9(B)-8, Table 3.9(B)-9 and Table 3.9(B)-10 are contained in Table 3.9(B)-11.

4. For those piping systems which are non-ASME III, design criteria were specified so that structural integrity of such systems could be maintained during the most adverse plant condition.

For any of the above piping systems which contained in-line components, such as valves, flow elements, strainers, etc., the loads imposed on such items by the piping were verified to be less than the limits established by the vendor. If operators were included on such valves, the seismic accelerations imposed were verified to be less than the levels to which the unit was qualified, either on a structural integrity criteria for nonactive valves or on an operability criteria for active valves. The latter criteria are discussed further in Subsection 3.9(B).3.2.

Analyses of all seismic Category I piping systems have been conducted using either the ADLPIPE, ADLPIPE-2 or IMAPS computer program. In each of these programs, the mathematical models employed to represent the piping system consisted of lumped masses interconnected by beam elements whose elastic properties matched those of either the piping section, or an in-line component, such as a valve. Support elements and equipment attachment points were included in such models. Lumped masses, offset from a section centerline, were included, when necessary, to model valve operators.

Structural boundaries of a mathematical model were defined by equipment connections, by support system anchors or by restraints which formed the boundary between seismic Category I and nonseismic Category I piping systems.

3.9(B).3.2 Pump and Valve Operability Assurance

The pumps and valves identified as active, ASME Section III components that must perform a mechanical motion during the course of performing their safety function in shutting down the plant to and maintaining it in a cold shutdown or in mitigating the consequences of a postulated event, are listed in Table 3.9(B)-26 and Table 3.9(B)-27, respectively. These active pumps and valves are classified as seismic Category I, and are designed to perform their intended functions during the life of the plant under all postulated plant conditions. The operability of these active pumps and valves is assured by adherence to the design limits and supplemental stress requirements specified in NRC Regulatory Guide 1.48.

Safety-related active valves are qualified by prototype testing and analysis; safety-related active pumps are qualified by analysis and functional test. All applicable loads, such as seismic, nozzle and operating loads are considered in the test program and the analysis. Operational tests of the originally-installed components at design basis conditions were performed during plant test start-up. All originally-designated active valves were tested by vendor at full pressure, nozzle loads, and seismic loads during the operability tests. Full flow conditions were generally not included during these tests but a differential pressure was applied and was effective during the opening stage of the valves. Except for the main steam isolation valve and the feedwater isolation valve, the other valves were not tested at elevated temperatures. ANSI B16.41 provided the technical guidance for qualifying equipment by similarity analysis and prototype testing. Where components are a part of a pressurized piece of equipment within the pressure boundary, testing of the assembly was performed. Components not originally-designated as active were reviewed for loading and seismic accelerations, to ensure that they meet active component requirements. By means of these programs, the structural integrity and the ability to perform the safety-related functions are assured for the active components during the postulated plant loadings. The details of the operability assurance programs are presented below.

a. Pump Operability Assurance Program

The overall operability verification program consists of both testing and analysis. The test programs used to establish the operability of all active seismic Category I pumps include tests by the pump manufacturer prior to installation, and by the licensee after installation in the plant. The tests performed at the pump manufacturer's facility include hydrostatic tests of pressure retaining parts in accordance with the requirements of ASME Section III, Subsection NC or ND, and pump hydraulic performance tests in accordance with Hydraulic Institute standards. After the pump is installed in the plant, the pump is subjected to pre-operational tests, including cold hydro tests and hot functional tests. The required periodic in-service inspection and operation tests, in accordance with the IST Program or other procedures, can further demonstrate reliability of the active pump for the design life of the plant.

In addition to these tests, seismic Category I active pumps are analyzed for operability when subjected to the SSE seismic condition and applicable or conservative seismic nozzle loads, to ensure the pump will not be damaged during the seismic event and that the pump will continue operating after the SSE seismic condition. Three areas of analysis are performed on the pumps: the pump body, the rotating assembly and the pump supports.

In the pressure boundary pump body analysis, the stress limits for all loading combinations, including faulted, are given in Table 3.9(B)-12. Table 3.9(B)-26 lists all AE-supplied active pumps. The rotating assembly is analyzed for operability during the faulted condition by assuring that: (1) the deflection of the pump impeller shaft will not exceed the clearance between the impeller and impeller casing and (2) the bearing will not be subjected to excessive loads imposed by deflection of rotating assembly and by differential movement of the coupling between the pump and pump driver shaft. The pump supports, including the base frame and anchor bolts, are analyzed for dead weight, nozzle loads, operating loads and seismic loads. The stress limits for the supports are those of AISC Manual of Steel Construction, and are described in Table 3.9(B)-12.

The lowest natural frequency of all active pumps, except the service water pumps, is demonstrated by test or analysis to be greater than 33 Hz. The service water pump is a long deep well pump with a natural frequency of 9.7. The pump has two lateral restraints which maintain lateral displacements within the limits of the available clearances. Additionally, all stresses are limited to 1.5 S, thereby assuring that the pump operability is maintained in the faulted loading conditions. Pumps having a natural frequency above 33 Hz, are considered to be rigid, and the problems with amplification between the component and structure are avoided.

To avoid damage during the faulted plant condition, three areas of analysis are performed on the motor: the supports, the rotor assembly, and the motor stator frame. The supports, bolts, and the stator frame are analyzed for deadweight, operating loads, and seismic loads and the stress limits are those of the AISC Manual of Steel Construction. Deflection of the rotor shaft was compared to the clearance between the stator and the rotor, to ensure that a rubbing-type failure will not occur. The angular and parallel shaft deflections at the coupling were calculated and compared to the allowables for the coupling. Rotor shaft stresses and bearing loads were evaluated and compared to allowables for the faulted plant conditions. All motor stresses are limited to the region of elastic deformation of the material stress-strain relationship and thereby provide assurance that operability is maintained in the faulted condition.

b. <u>Valve Operability Assurance Program</u>

The operability assurance program for seismic Category I active valves of all pipe sizes is comprised of tests and analysis. This program provides assurance that these valves will perform their mechanical function in conjunction with a design basis accident during a seismic event. The active valves are subjected to several tests prior to installation; namely, a shell hydrostatic test to ASME Section III requirements, seat and disc hydrostatic tests, and functional tests. After installation, preoperational tests are performed. Periodic in-service inspections and periodic in-service testing, in accordance with the IST Program, or other Station procedures, or Technical Specification surveillances, further verify and assure the functional ability of the safety-related active valves.

The valve body and other pressure retaining parts of active valves are designed and analyzed by considering operating loads and seismic-induced nozzle loads. For valves with extended structures, an analysis of the extended structure is performed applying static, equivalent seismic loads of 3g for each of the three principal axes acting at the center of gravity of the extended structures. The maximum allowable stress limits applied in these analyses demonstrate structural integrity and compliance with the limits specified by the ASME Section III Code for the particular ASME Class of valve analyzed. Stress limits for all loading combinations are presented in Table 3.9(B)-13 for Class 2 and 3 safety-related active valves and Table 3.9(B)-14 for Class 1 safety-related active valves. Table 3.9(B)-27 lists all AE-supplied active valves.

The valve body for active and nonactive valves is qualified by analysis and accounts for the interface loading imposed by the actuators.

The valve actuators for active and nonactive valves are qualified by tests in accordance with IEEE 323-1974 and IEEE 344-1971. However, a 1.5 factor is applied to the sinusoidal input motion and, therefore, compliance with IEEE 344-1975 is achieved. The response of equipment at the resonant frequency at 5 percent damping for a continuous sinusoidal input is amplified approximately ten times, compared with an amplification of three times for a random motion input. By applying a factor of 1.5 to the sinusoidal input, the comparative response is 10/3 x 1.5 = 5 to 1, which is conservative.

In addition to the above functional tests and analyses, representative originally-designated active valves of each design type with overhanging structures were tested to verify operability during simulated seismic events by demonstrating operational capabilities within the specified limits.

Functional specifications for originally-designated active valve assemblies were not prepared, but the requirements of R.G. 1.148 were contained within the design specifications and system specifications. The requirements of ANSI B16.41 were not part of a specific test program but all of the individual tests defined, except for the vibration endurance tests, were performed as part of the series of tests comprised of vendor hydro tests and seismic tests and plant startup testing. The valve(s) chosen for the parent valve(s) for vendor seismic testing generally complied with the size extension limitations of 200 percent to 50 percent except as follows:

(1) Posi-Seal butterfly valves, Class 2 and 3, 150 lb. carbon steel body with matrix operator, sizes 14 to 36 inches, were qualified by tests performed on a 30-inch valve.

(2) Walworth gate valves, Class 2 and 3, 150 lb. carbon steel body with Limitorque operator, sizes 3 to 16 inches, were qualified by tests performed on 8 and 16 inch valves. Although the 3-inch valve is below the 50 percent criteria, evaluation of the valve dimensions indicate sufficient conservatism so that operability is assured.

The testing was conducted on a representative number of valves. Valves from each of the design types were tested, and the valve sizes which cover the range of sizes in-service were qualified by the tests. The test results were used to qualify all valves within the intermediate range of the installed sizes. The testing procedures and basic requirements for the operability verification of originally-designated active valves with overhanging structures are described below:

1. The valves are designed to have a fundamental frequency which is greater than 33 Hz. This is shown by suitable test or analysis.

2. The actuator and yoke of the valve system were statically loaded to seismic loads equivalent to 3g for each of the three principal axes. The static loads were applied at the center of gravity of the extended structure along the direction of the weakest axis of the yoke if the two horizontal seismic loads are combined as one static load input. The design pressure of the valve was simultaneously applied to the valve during the static deflection tests. The design pressure of the valve and the six components of specified nozzle loads were simultaneously applied to the static deflection tests.

3. The valve was then operated while in the deflected position, i.e., from the normal operating mode to the faulted operating mode. The valve had to perform its safety-related function within the specified time limits.

4. Alternatively, the operability verification of the valve, including the actuator and all other appurtenances, was demonstrated by satisfying applicable vibration test requirements.

5. Motor operators and other electrical appurtenances necessary for valve operation are qualified as operable during the seismic event by testing and/or analysis in accordance with the requirements of IEEE 323 prior to installation on the valve.

Since the accelerations used for the static valve qualification are 3.0g horizontal and vertical, the piping designer maintains the motor operator accelerations (or offset mass acceleration) to these levels with an adequate margin of safety.

If the frequency of the valve, determined by test or analysis, is less than 33 Hz, an analysis of the valve is performed to determine the acceleration value applicable to the extended portion of the valve. The analysis provides the amplification of the input acceleration considering the natural frequency of the valve and piping along with the applicable floor response spectra. The computed accelerations at the extended portion of the valve are then compared with the 3g values used in the static load tests or the shake table test results outlined in step 4 above to assure the operability of these active valves for the applicable seismic loads.

The structural integrity and operability verification of the seismic Category I active check valves are ensured by a combination of the following tests and analysis:

1. Stress analysis including the applicable seismic loads for valve parts that could affect the operability of the valve

2. In-shop hydrostatic test and in-shop seat leakage test

3. Periodic in situ valve inspection to assure the functional ability of the valve.

The seismic Category I safety relief valves are subjected to stress analyses for parts that could affect the operability of the valve for the applicable seismic loads and operating loads. These valves are also subjected to in-shop hydrostatic test, seat leakage test and periodic in situ valve inspection. In addition to these tests, a representative number of originally-designated active valves were qualified by shake table test. Random seismic input is applied to the valve and the pressure is increased until the valve mechanism actuates. Successful actuation within the design requirements of the valve is demonstrated to assure the functional capability of the valve.

c. Operability Assurance Program Results for Active Pumps

1. Pumps

(a) Primary Component Cooling Water Pumps

The structural integrity and operability of the primary component cooling water pumps have been demonstrated by analytical method. The analysis was performed by using the ASME III 1974 Code, AISC criteria (support only), plus the equipment specification. The pumps were analyzed with a combined load consisting of deadweight, seismic loads and nozzle loads. Ingersoll-Rand Co. Report Number EAS-TR-7535N shows that the 14x23-5 pump satisfies all of the applicable structural integrity and operability requirements of ASME Boiler and Pressure Vessel Code, Section III, and Regulatory Guide 1.48.

A lumped mass dynamic model was used with the ANSYS Computer Program to determine the fundamental natural frequencies. The calculated first mode natural frequency is 46.19 Hz, so the pump assembly is essentially rigid. A static analysis was performed to determine the stresses and deflections which will result from the application of the actual plant suction and discharge nozzle loads in conjunction with internal pressure, deadweight, and seismic acceleration loads. The resulting stresses, loads and clearance requirements are summarized on Table 3.9(B)-15.

To complete the operability assurance program, the Westinghouse-supplied motor was qualified for Class 1E electric equipment, as defined in IEEE 323-1971 and IEEE 344-1971. The motor was analyzed using AISC criteria with a combined load consisting of deadweight, torque and seismic loads.

(b) Containment Spray Pumps

An operability assurance analysis was provided for the Bingham-Willamette 6x10x14B - CD containment spray pump-motor set. The applicable codes are ASME III, 1974, AISC criteria (support only), and the equipment specification. The pump and motor set was analyzed for combined loading consisting of deadweight, seismic loads, and nozzle loads. In addition, the operating torque was considered in the shaft analysis. The natural frequencies of both the containment spray pump and the motor were computed, and all frequencies were found to be greater than 33 Hz; therefore, the pump assembly is considered essentially rigid.

A static analysis was used to determine the stresses and deflections which result from the application of nozzle loads, deadweight and seismic loads. A summary of the deflections and stresses is given in Table 3.9(B)-16, which shows the stresses and deflections are well within allowable limits. The deflection of the pump shaft was compared to the minimum clearance. The motor rotor deflection was compared to the air gap. The coupling deflection and the angular misalignment were compared to the vendor-specified allowable limits.

The Westinghouse-supplied motor was analyzed using AISC criteria, with a combined load consisting of deadweight, torque and seismic loads.

(c) Emergency Feedwater Pumps

Detailed seismic analyses were performed for the two emergency feedwater pumps, one of which uses an Ingersoll-Rand Model 4x9 NH-10 turbine-driven pump and the other, a model 4x9 NH-10 motor-driven pump. The pumps were each analyzed for a combined load consisting of deadweight, seismic loads and nozzle loads. Ingersoll-Rand Co. Report Numbers EAS-TR-8001 and EAS-TR-7925 established that both pumps were found to satisfy all of the applicable structural integrity and operability requirements of the ASME III Code, AISC criteria (support only), Regulatory Guide 1.48 and equipment specification.

A mathematical model was developed and dynamic frequency analysis was performed using the computer code ANSYS. The calculated first mode natural frequency is 48 Hz for the turbine-driven pump assembly and 49.3 Hz for the motor-driven pump set. The two pump assemblies were considered to be rigid, and static analysis was used to determine the structural responses. The resulting critical deflections versus clearance requirements are summarized in Table 3.9(B)-17.

Testing and analysis was performed to qualify the GS-2N Terry Steam Turbine which is used to drive the 4x9 NH-10 pump. The Turbine System was tested at Wyle Laboratory using random motion test. Supplementary analysis was performed for the pressure-retaining parts and other essential components of the steam turbine.

The motor was analyzed using AISC criteria for a combined load consisting of deadweight, torque and seismic loads. Westinghouse Seismic Qualification included in Ingersoll-Rand Co. Report #75F32233 established that the structural integrity and operability are satisfied.

(d) Service Water Pump

Seismic analyses of the service water pump (Johnston Pump Company Model 30 DC, 2 stage vertical pump) were performed to verify operability and ability to withstand the postulated seismic events. A multi-degree-of-freedom lumped mass model was developed to represent the pump-column assembly and the discharge head-motor combination. The model consists of a series of mass points connected by weightless two-dimensional beams between nodal points. The shaft assembly between the motor and bowls is supported at each bearing along the column, and linear springs were used to represent the bearing stiffness. The vertical flexibility of the base plate was represented by a linear spring, and a torsional spring was used to represent the spring rate of the base plate subjected to the applied moment. The natural frequencies, deflections, loads, and stresses were obtained with the aid of computer program ANSYS.

The calculated fundamental frequency of the pump is 9.613 Hz for the entire structure. The deflection of the impeller relative to the pump casing was determined by analysis to be 0.002 in. A deflection of .002 inch translates into a natural frequency of 70 Hz. The pump is, therefore, considered rigid and per FSAR commitment, seismic operability testing is not required. A dynamic analysis was performed to determine the structural responses due to the seismic loads for the vertical and lateral directions. The nozzle loads and other normal operating loads were combined with the seismic loads.

A supplemental analysis was performed by Ingersoll-Dresser Pump for the IDP Model 42APK service water pump, which utilizes the previously existing discharge head, motor, and baseplate (IDP Technical Report 2000-17). The resulting stresses were compared to the allowable stresses, and the resulting deflections were compared to operating clearances or other operating criteria. A summary of the stresses and deflections is given in Table 3.9(B)-18.

The seismic, stress and deflection analysis of the General Electric Company's 600 HP, Model 5K6338XC102A, pump motor was performed by McDonald Engineering Analysis Company. The analysis was directed toward proving both the structural integrity and functional capability of the motor. The seismic loads, including the reduced magnetic and centrifugal loads, were imposed on the mathematical model of the motor assembly, and the resulting stresses and deflections were compared to the code allowables and the operating clearances or other limiting criteria. McDonald Engineering Company's Report, ME 267, established that the motor satisfies all of the applicable requirements.

(e) Cooling Tower Pump

The seismic analysis of the cooling tower pump was performed for the Model 33NLC Johnston vertical pumps. The analytical approach was essentially the same as for the service water pump. Johnston Seismic Report ME-772 shows that the pump satisfies all of the applicable structural integrity and operability requirements of ASME III and the specification requirements.

The deflection of the impeller relative to the pump casing was determined by analysis to be 0.0019 in. The pump is, therefore, considered rigid and per FSAR commitment, seismic operability testing is not required.

A dynamic analysis was performed to determine the stresses and deflections which will result from the application of the seismic loads, deadweight and nozzle loads. A supplemental analysis was performed by Ingersoll-Dresser Pump for the IDP Model 29LKX cooling tower service water pump, which utilizes the previously existing discharge head, motor, and baseplate (lDP Technical Report 2000-16). The resulting stresses, loads and clearance requirements are summarized in Table 3.9(B)-19.

The General Electric-supplied 800 HP Model 5-K 6339XC179A motor was analyzed by McDonald Engineering Analysis Company.

It is similar to the motor used for service water pump. Stress Report #266 established that the motor satisfies all of the applicable requirements.

(f) Diesel Fuel Oil Transfer Pump

Operability of the diesel fuel oil transfer pump under the most adverse applicable combination of normal loads, nozzle loads, and seismic loads has been analytically demonstrated by the pump manufacturer, Delaval IMO Pump Division. The pump assembly consists of a Delaval IMO screw-type pump (N3DBS-187) and a Westinghouse electric motor mounted on a common bedplate.

The natural frequency of the unit was calculated; the fundamental natural frequency was determined as 228 Hz. Static analysis was used to determine the structural responses and the resulting stresses; deflections are summarized in Table 3.9(B)-20.

The Westinghouse supplied 2 HP, Type T, fan-cooled AC motor was analyzed by Westinghouse Medium Motor & Gearing Division. Westinghouse Qualification Document MM-9112 and Certification letter dated February 13, 1981 assure that the motor satisfies all of the applicable requirements. The seismic analysis report is available for review at the Westinghouse Medium Motor & Gearing Division.

(g) Spent Fuel Pool Cooling Pumps

A detailed seismic analysis was performed for the Model 6X8X12 CF spent fuel pool cooling pumps supplied by Bingham-Willamette. This analysis verified the equipment's ability to resist the design basis seismic event by demonstrating the adequacy of critical components when subjected to normal and seismic loads. The applicable design codes were ASME Section III 1974 edition for pressure boundary components and AISC 1970 edition for structural support members. Loads considered in this analysis include nozzle, motor torque and seismic inertia. The natural frequencies of the pump and motor were calculated and all frequencies were found to be significantly greater than 33 Hz, refer to Table 3.9(B)-29. Therefore, the spent fuel pump/motor assembly is considered rigid.

The stresses and deflections for critical components of the pump/assembly resulting from the application of nozzle, deadweight, seismic inertia and motor torque loads as appropriate were determined using static methods. Critical components addressed include the shaft, pedestal weld and the pump, motor and assembly hold down bolts. Additionally, deflections of critical rotating components were also calculated and compared to allowable clearances. For example, it was demonstrated that the motor rotor deflection was less than the available air gap between the rotor and stator. Likewise, the shaft coupling deflections and possible misalignment was confirmed to be within acceptable limits. The pump motor manufacturer, Westinghouse, has also performed a seismic analysis demonstrating acceptability to the specified seismic requirements.

d. Operability Assurance Program Results for Active Valves

The results of seismic tests and analysis of originally-designated active valves was provided in the document entitled, "Public Service Company of New Hampshire, Seabrook Station Units 1 & 2, Seismic Qualification Review Team (SQRT) Equipment List," which was forwarded to Mr. Frank J. Miraglia, Chief Licensing Branch #3, Division of Licensing, under cover of PSNH's letter, dated May 27, 1982.

Valves that are not ASME Section III or are not 1E that perform a mechanical motion to accomplish or support a safety function* are periodically tested per the IST Program, or other station procedures, or Technical Specification surveillances. These valves (both BOP and NSSS) are listed in Table 3.9(B)-28. Valves located in a harsh environment have been analyzed for environmental qualifications and are included in the associated station programs.

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* This includes some NNS valves, located in portions of systems that serve no safety function. Per the NRC safety evaluation for Seabrook Station (NUREG 0896), the NNS main feedwater regulating and bypass valves provide for redundant feedwater line isolation function to mitigate a main steamline or feedwater line break. NURGE 0896 also indicates that these fail-closed valves serve as an acceptable backup to the main feedwater isolation valves. In response to NRC Generic Letter 96-06, it was identified that several relief valves in NNS piping protect containment penetration isolation valves from an overpressure condition. Subsequently, it was identified that a relief valve in a normally isolated NNS piping section protect ECCS boundary isolation valves from an overpressure condition. To ensure continue reliability of the above NNS relief valves, they are being periodically tested.

3.9(B).3.3 Design and Installation Details for Mounting of Pressure Relief Devices

The installation and design of pressure relief devices comply with the rules of ASME III, Paragraph NB-7000, and NRC Regulatory Guide 1.67.

a. Overpressure Protection for Reactor Coolant Pressure Boundary (RCPB)

The pressurizer in the Reactor Coolant System is provided with three safety valves and two power-operated relief valves for overpressure protection. These valves discharge through a closed piping system to the pressurizer relief tank, where the steam is condensed and cooled by mixing with water. The piping system and supports are designed to satisfy the following design criteria:

1. Stress limits for load combinations listed in Table 3.9(B)-8 for safety Class 1 piping from the pressurizer to the safety and relief valves
2. Stress limits for load combinations listed in Table 3.9(B)-9 for nonsafety class piping downstream of the safety and relief valves to the pressurizer relief tank
3. Load limits on pressurizer vessel nozzles as established by the manufacturer of the pressurizer vessel
4. Load limits on valve connections as established by the manufacturer of the valves.

The three safety valves are mounted on the pressurizer nozzles with the short inlet pipe and elbow necessary to position the valves vertically. The total length of pipe, elbow and weld-neck flange is approximately 24 inches and is as short as possible to minimize the pressure drop on the inlet side of the valve.

When the valves open, the dynamic effects from the flow of water and steam are included in the design analysis.

These transient load effects on the piping system, upstream and downstream of the safety and relief valves, have been evaluated in the following manner:

1. Safety Valve Piping System

A static analysis was performed for the Safety Valve Piping System in which the peak transient loads obtained from a RELAP 5 analysis and multiplied by a dynamic load factor (DLF) were applied. The Pressurizer Safety Valve Piping System contains no water seals nor is subjected to water slugs.

2. Pressurizer Relief Valve Piping System

Both static and time history analyses were performed for the Pressurizer Relief Valve Piping System using transient loads obtained from a RELAP 5 analysis. The Pressurizer Relief Valve Piping System contains water seals and is subjected to water slugs. The effects of these two items were fully accounted for in the RELAP 5 analysis.

The RELAP 5 computer code, (Reference 1), was used to generate thermal hydraulic characteristics of the flow along the piping system, from which tables of the wave force versus time for each leg have been derived. To evaluate piping stresses and support loads, the maximum force for each leg has been selected and applied statically to the piping system in the most conservative fashion using a dynamic load factor (DLF) based on the valve opening time and the system dynamic characteristics. In cases where time history analyses were performed, the appropriate thermal hydraulic forcing function was applied to the applicable pipe segment. The developed stresses and loads on nozzles were combined with the other applicable loads from Table 3.9(B)-8 and Table 3.9(B)-9. These were compared with the allowable stresses and allowable nozzle loads. The simultaneous discharge from all valves has been assumed in the thrust analyses.

b. Overpressure Protection for the Secondary (Main Steam) System

A multiple-valve installation, comprised of five safety valves, is provided in each of the four main steam lines. The valves are installed on main steam piping headers, outside of the Containment Building in a piping chase between the containment penetration and the main steam isolation valves. The safety valve discharge side is configured to minimize reaction forces at the valve branch/main header intersection point. The vertical branch line from the main steam piping header to each individual valve has a forged flange and sweepolet welded to the header. Safety valves are bolted directly to the flanges.

The effect of the valve discharge transient was obtained by static application of an assumed discharge force, as obtained from the valve manufacturer, with a dynamic load factor DLF based on the system dynamic characteristics. It has been assumed that all five valves discharge simultaneously. The system of piping supports and rigid restraints limits both dynamic and static loadings to the piping system to code allowable stresses for the load combinations listed in Table 3.9(B)-9.

c. Safety and Relief Valves for Various Auxiliary Systems

Mounting of safety and relief valves on auxiliary piping systems uses standard piping components: flanges, buttwelded or socket-welded tees, weldolets® and sockolets® for pipe branches to the valves. The valves and valve discharge piping utilize flanged joints, buttwelded and socketwelded connections. Branch connections are qualified using code standard calculations for tees with proper intensification factor (ASME III, Table NB-3682.21 or NC-3652-4). The alternative method for branch qualification is the Bijlaard method using the SPHNOZ/CYLNOZ computer program. The load combination for calculating stresses is according to Table 3.9(B)-9. These were compared with the allowable stresses.

The following basic installation of safety valves outlet piping has been utilized:

1. Open discharge with the minimum piping length, or no piping attached to the valve discharge and discharging to atmosphere.

2. Open discharge system, discharging directly to atmosphere through individual piping systems, or common header combining discharge from several valves.

3. Closed discharge system, discharging to a container through an individual piping system, or common header combining discharge from several valves.

The discharge reaction forces have been obtained from one of the following sources:

1. Valve manufacturer

2. For open steam discharge, from nonmandatory Appendix "O" of ASME III

3. For open and closed discharge systems with piping system connected to lve discharge, from UE&C proprietary computer programs MADIS, VALCLO, ELBFOR, and WATER (described in Subsection 3.9(B).1.2).

The stress analyses of piping systems downstream and upstream of a valve have been obtained by applying the reaction forces statically with a dynamic load factor (DLF) as appropriate based on the system dynamic characteristics. Piping stresses from the safety valve discharge have been combined with the stresses from other applicable loads in accordance with the load combination from Table 3.9(B)-9. These were compared with the allowable stresses.

d. Results of Analysis

The results of a dynamic analysis of the various piping systems have been summarized in the tabular format represented by Table 3.9(B)7-21 and Table 3.9(B)-22.

3.9(B).3.4 Component Supports

a. ASME Code Class 1, 2 and 3 Piping Supports

1. Jurisdictional Boundaries

The jurisdictional boundaries of supports designed and fabricated to Subsection NF requirements as shown in NF-1000 for plates, welding and bolting is as follows:

(a) Plates

(1) Support plates that are embedded in concrete with integral embedded anchors (studs) do not fall within NF jurisdiction, whether or not they extend beyond the surface of the concrete.

(2) Loose or adjustable base plates which support only compressive loads do not fall within NF jurisdiction.

(3) Loose plates that are welded to component supports, such as surface-mounted plates, fall within NF jurisdiction.

(b) Welding

Welds used to attach NF supports to building steel, supplementary steel, or intervening members fall within NF jurisdiction.

(c) Bolting

(1) Embedded custom-designed anchor bolts are designed and fabricated to AISC requirements and also to the additional materials, certification and NDE requirements of Subsection NF.

(2) Expansion anchors which are manufactured and stocked as catalogue items do not fall within NF jurisdiction.

2. Loads

The loads (and movements of components) considered in the analysis of ASME Code Class 1, 2 and 3 piping supports are shown in Table 3.9(B)-23. The loads were combined to determine a worst case in each direction for normal and upset condition and faulted condition. Friction forces due to thermal movements of the piping bearing on the support structures were considered by calculating a friction force using a coefficient of friction of 0.35, or applying the thermal displacement to the support structure in those cases where the support structure is flexible.

3. Types of Supports, Design and Service Limits

(a) Linear Type Supports

Stresses were calculated using linear elastic analysis in accordance with ASME Section III, Division 1, Subsection NF. The maximum allowable stresses used were:

(1) For normal and upset load conditions, the allowable stresses were in accordance with ASME, Section III, Division 1, Subsection NF, and Appendix XIII.

(2) For the faulted load condition, the allowable stresses used were in accordance with the ASME Code, Appendix F and USNRC Regulatory Guide 1.124 for essential systems.

(3) The stiffness values used for support design were:

Pipe size (OD, in.)	Stiffness (lb./in.)
Up to 2½	$1x10^4$
2½ to 6	$1x10^5$
Above 6	$1x10^6$

In those cases where the support stiffness was less than that specified above, the piping analysis was reviewed to determine the impact on the component.

(4) Component supports are designed to be in the rigid range (natural frequency fn≥33 Hz). In cases where the frequency is less than 33 Hz, the analysis of the piping system was reviewed to assure that the piping analysis remained valid.

(5) The thermal movement of the component at the support was accommodated through clearance included in the component support design.

(6) Component supports are connected to concrete walls and slabs by either welding to embedded plates, or by bolting to the concrete with either concrete expansion anchors (wedge type) or concrete inserts. The response to the NRC's IE Bulletin No. 79-02, (Reference 2), was used as a guide for the design of the concrete expansion anchors. The maximum allowable design loads for the concrete expansion anchors for ASME Class 1, 2, and 3 supports were developed using the manufacturer's ultimate loads and a safety factor of 4 for worst case loading (normal and upset or faulted loads). Embedded plates, expansion anchors, and concrete inserts installed in concrete degraded by ASR provide full structural capacity up to the ASR expansion level defined in Table 3.8-18.

Baseplate flexibility and shear-tension interaction were accounted for in the design of the concrete expansion anchors.

(7) Lug attachments welded to Class 2 and 3 pipes are qualified by a procedure whose methodology is equivalent to, but more conservative than, that presented in Code Case N-318-1.

Local stress levels in the pipe resulting from applied lug loads are obtained by multiplying the nominal stress in the lug at the lug/pipe interface by the appropriate B or C index (as defined in Code Case N-318-1) for each individual loading condition. The local stresses are superimposed upon the general pipe stress as determined from program ADLPIPE to establish the total stress level in the pipe for that loading condition.

Loading conditions required to be considered for Plant Normal, Plant Upset, Plant Emergency, and Plant Faulted Operating Condition are defined (per appropriate Updated FSAR section), and total stress in the pipe is obtained from summing the stresses for each individual loading condition that must be considered.

Local stress levels determined using B indices are added to the general stress levels from ADLPIPE and this sum is compared against allowable limits to demonstrate structural integrity. For the pipe wall, local stress levels determined using C indices are added to the general stress levels from ADLPIPE, and this sum is compared against the allowable range of stress (S_h+S_a).

(b) <u>Component Standard Supports</u>

Component standard supports were selected on the basis of:

(1) Maximum design loads (and movements) as established in Table 3.9(B)-23

(2) Allowable loads (and movements) provided by manufacturers' component standard support catalog load capacity data sheets

The allowable loads were certified by the manufacturer to be generated in accordance with ASME Section III, Subsection NF, for all load conditions.

4. ASME Code Class 2 and 3 Support Inspection

All Class 2 and 3 supports will receive surveillance and audit inspection by the Authorized Nuclear Inspection Agency (ANIA) in lieu of in-line inspection hold points by the Authorized Nuclear Inspector (ANI). This modified program will be applied to all ASME Class 2 and 3 pipe supports and I&C supports of the following systems:

AS - Auxiliary Steam

CAH - Containment Air Handling

CAP - Containment Air Purge

CBS - Containment Building Spray

CC - Component Cooling Water Primary

CGC - Combustible Gas Control

CO - Condensate

COP - Containment Online Purge

CS - Chemical and Volume Control

DG - Diesel Generator

DM - Demineralized Water

FP - Fire Protection

FW - Feedwater

LD - Leak Detection

MS - Main Steam

MSD - Main Steam Drains

NG	-	Nitrogen Gas
RC	-	Reactor Coolant
RH	-	Residual Heat Removal
RMW	-	Reactor Makeup Water
RS	-	Resin Sluicing
SS	-	Sample System
SA	-	Service Air
SB	-	Steam Generator Blowdown
SF	-	Spent Fuel Pool Cooling
SI	-	Safety Injection (Emergency Core Cooling)
SW	-	Service Water
VG	-	Vents
WG	-	Waste Processing - Gaseous
WLD	-	Waste Processing - Liquid Drains

In addition, all Class 2 and 3 supports included in the following systems will receive additional surveillance inspection (i.e., increased frequency of surveillance inspection):

MS - Main Steam

FW - Feedwater

CS - Chemical and Volume Control

SB - Steam Generator Blowdown

SW - Service Water

SI - Emergency Core Cooling

RH - Residual Heat Removal

5. ASME Code Class 2 and 3 Support Documentation

Code Symbol Stamping or executing NF-2 Data Reports will not be required for all Class 2 and 3 pipe supports and I&C supports. Instead, a statement will be attached to N-5 Data Reports indicating that design, procurement, fabrication, installation and examination have been performed in accordance with ASME III, Subsection NF, and confirmed by ANI surveillance and audit inspection.

b. ASME Code Class 1, 2 and 3 Valve Supports

ASME Code Class 1, 2 and 3 valves, including active valves, are generally supported by the pipe attached. Exceptions to this method of support occur where the valve has an excessively heavy actuator relative to the valve body weight, and/or where the pipe line size is too small to provide adequate support, and where flexible (less than 33 Hz) pneumatic operators are used. For these exceptions, the analysis and design of the supports are in accordance with the preceding subsection on piping supports.

c. Mechanical Equipment Supports

ASME Code Class 2 and 3 component supports, including supports for pumps, vessels, tanks and heat exchangers, are designed to withstand the loads resulting from the loading criteria and operating condition categories as defined in Subsection 3.9(B).3.4. The allowable stresses defined in AISC "Manual of Steel Construction" were used for plant conditions associated with OBE. For plant conditions associated with SSE, the stresses were limited to 90 percent of yield stress for the material involved, or in accordance with Subsection NF of the ASME Boiler and Pressure Vessel Code, Section III. The requirements for adequacy of active Class 2 and 3 pump supports require that both stress analysis and an evaluation of pump/motor support misalignment be performed.

The supports of certain mechanical equipment purchased circa 1974 were designed in accordance with the requirements defined in the AISC Manual of Steel Construction. In addition, the following criteria were included in the support designs:

1. Material properties used in conjunction with the support design were obtained from the tables for material strength values in the ASME III, Subsection NA, Appendix I.

2. The allowable bolt stresses were derived from the AISC Specification, without use of one-third increase factor for Normal and Upset Conditions. For the faulted condition the AISC allowable of 0.6 Fy was multiplied by the strength factors noted in SRP 3.8.3 and 3.8.4.

3. The loading considered in the design of the supports and anchor bolts are the same as those imposed on the components. More specifically, the appropriate loads are applied to the components and the resulting reactions are used to design the supports.

4. For the faulted condition, tensile and bending stresses were limited to 90 percent of the material yield strength and shear stresses were limited to 60 percent of the material yield strength which compare favorably with the limits defined by ASME III, Subsection NF, for faulted conditions.

5. Buckling evaluations were performed in accordance with the AISC criteria without use of increase factor for faulted conditions.

6. The highest value of KL/R is less than 20 for all mechanical components (excluding piping systems).

The following tabulation shows the stress limits used for various bolt materials:

1. Stress Limits for Anchor Bolts for Equipment

Bolt Material	Allowable Tensile Stress	Allowable Shear Stress
ASTM A193 Grade B7		
Under 2-1/2"∅		
Fy = 105 ksi	Ft = 0.6 Fy 0.5 Fu	Fv = 0.4 Fy
Fu = 125 ksi	= 62.5 ksi	= 42 ksi
ASTM A540 Grade B23 Class 4		
Up to 3"∅		
Fy = 120 ksi	Ft = 0.6 Fy 0.5 Fu	Fv = 0.4 Fy
Fu = 135 ksi	= 67.5 ksi	= 48 ksi
ASTM A354 Grade BD		
For 1/4" to 2-1/2"∅		
Fy = 130 ksi	Ft = 0.6 Fy 0.5 Fu	Fv = 0.4 Fy
Fu = 150 ksi	= 75 ksi	= 52 ksi

2. High strength bolts for equipment on structural steel and for steel-to-steel connections.

ASTM A325

1/2" to 1"∅	1-1/8" to 1-1/2"∅
Fy = 92 ksi	Fy = 81 ksi
Fu = 120 ksi	Fu = 105 ksi

ASTM A490

1/2" to 1-1/2"∅

Fy = 130 ksi

Fu = 150 ksi

All allowable tension and shear values are in accordance with Manual of Steel Construction AISC.

For the faulted condition, the strength factors of 1.6 or 1.7 as noted in SRP 3.8.3 and 3.8.4 were applied to the above.

d. Snubbers

1. Design Criteria

Shock suppressors, mechanical and hydraulic, used in ASME Code Class 1, 2 and 3 piping or equipment conform to the requirements of the ASME B & PV Code Section III, Subsection NF.

The snubbers' intended use is as shock arrestors only, and meet the following guidelines:

(a) Design life of forty years

(b) Operate under environmental conditions described in the Service Environment Chart, Updated FSAR Figure 3.11-1

(c) Applicable codes are as follows:

(1) Mechanical Shock Suppressors - Summer Addenda of the 1974 B & PV Code.

(2) Hydraulic Shock Suppressors - 1977 Edition, Winter, 1979 Addenda. Note: Later Editions of the Code may be used as provided by Section XI, Para. IWA-4223.

(d) DELETED

(e) DELETED

(f) Both mechanical and hydraulic units are designed in such a way that they will not form a rigid restraint.

(g) DELETED

(h) Hydraulic units are designed with a means for monitoring fluid levels in the reservoirs.

(i) DELETED

(j) Materials used for the fabrication of ASME Code parts shall conform to the applicable requirements of Article NF-2000 of the ASME B & PV Code.

(k) Fabrication of shock suppressors shall be in accordance with the applicable requirements of Article NF-4000 of the ASME B & PV Code.

(l) DELETED

(m) Load ratings shall be verified in accordance with the requirements of Article NF-3260 of the ASME B & PV Code.

(n) Each hydraulic snubber shall be tested in compression and tension to 10 percent of its rated load and checked for leakage of the hydraulic fluid. If fluid forms droplets, drips or runs off the piston rod, the shock suppressor shall be rejected.

(o) Shock suppressors' packaging shall be designed to protect against salt spray, rain, dust, water vapor, shock and vibration during shipping, handling and storage. Where possible, shock suppressors shall be packaged fully assembled in a single shipping container.

(p) Mechanical units shall be designed to operate normally between 50°F and 300°F.

Hydraulic units shall be designed to operate normally between 30°F and 140°F with temperature excursions up to a maximum of 300°F.

2. <u>Snubber Installation and Operability Verification</u>

(a) <u>Pre-Service Examination</u>

A pre-service examination will be made on all snubbers. This examination will be conducted during and after snubber installation and will, as a minimum, verify the following:

(1) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

(2) The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.), are according to design drawings and specifications.

(3) Small snubbers are not seized, frozen, or jammed by manual exercising during installation. Large snubbers (those that cannot be manually exercised) will be identified and examined for proper movement during preoperational testing as discussed in Subsection 3.9(B).3.4d.2(b) below.

(4) Adequate swing clearance is provided to allow snubber movement.

(5) If applicable, fluid is to the recommended level and is not leaking from the snubber system.

(6) Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire and cotter pins are installed correctly.

Prior to the performance of the thermal expansion test, an inspection of all listed snubbers covering items (1), (4) and (5) will be performed as a test prerequisite. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements will be repaired or replaced and re-examined in accordance with the above criteria.

(b) Pre-Operational Testing

During thermal expansion testing, snubber thermal movements for systems whose operating temperature exceeds 250°F will be verified as follows:

(1) During initial system heatup and cooldown, at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement.

(2) For those systems which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement.

(3) Verify the snubber swing clearance at specified heatup and cooldown intervals. Any discrepancies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval.

3.9(B).4 Control Rod Drive Systems

Refer to Subsection 3.9(N).4.

3.9(B).5 Reactor Pressure Vessel Internals

Refer to Subsection 3.9(N).5.

3.9(B).6 In-Service Testing of Pumps and Valves

An ongoing in-service test program will be provided to assure the operational readiness of certain Safety Class 1, 2 and 3 pumps and valves which perform a specific function in shutting down a reactor to a safe shutdown condition or in mitigating the consequences of an accident.

The in-service test program is based on the requirements given in the ASME OM Code, 2004 Edition and the requirements of 10 CFR 50.55a(f) except where specific written relief has been granted by the commission pursuant to 10 CFR Part 50, Section 50.55a(f)(6)(i). Applicability of future Code addenda will be as stated in 10 CFR 50.55a(f).

3.9(B).6.1 <u>In-Service Testing of Pumps</u>

In-service tests, analysis and record keeping will be performed for certain Code Class 1, 2 and 3 pumps in accordance with Subsection ISTB of the Code to assess pump operational readiness and to detect changes in pump hydraulic and mechanical performance relative to reference parameters. Reference values were established during pre-service testing and will be established after major maintenance or replacement.

Methods of measurement will be in accordance with ISTB-3500. Installed or portable instruments employed for measuring or observing test quantities will have accuracies equal to or better than that specified in Table ISTB-3510-1.

In-service test records to include test plans, documentation and required corrective action will be maintained in accordance with ISTB-9000.

A listing of Class 1, 2 and 3 pumps subject to in-service testing is provided in Station procedures. The Station procedures also specify the minimum test frequency, during plant operation, in which test quantities are to be measured, analyzed and documented. Plant personnel shall maintain test plans that include the type of hydraulic circuit normally used for testing.

3.9(B).6.2 In-Service Testing of Valves

In-service tests, analyses and record keeping will be performed for certain Code Class 1, 2 and 3 valves in accordance with Subsection ISTC of the Code to assess valve operational readiness.

The in-service testing program for valves is detailed in Station procedures. Each valve to be tested is identified by system, valve number, code class, type, function, category and applicable tests and test frequencies.

Each valve, prior to service, was tested as required by those tests defined for each valve in the In-service Test Program, in effect at that time. These pre-service tests were conducted under conditions similar to those to be experienced during subsequent in-service tests, to the maximum extent practicable.

When a valve or its control system has been replaced or repaired or has undergone maintenance that could affect its performance, and prior to the time it is returned to service, it will be tested to demonstrate that the performance parameters which could be affected by the replacement, repair, or maintenance are within acceptable limits.

Valves with remote position indicators, will be visually observed at least once every 2 years to verify that remote valve indications accurately reflect valve operation.

Valves which act as an isolation boundary between high pressure reactor coolant piping and adjacent low pressure systems, and whose undetected failure or degradation could lead to an inter-system LOCA, will be considered Category A or A-C valves and tested in accordance with this section and the Technical Specifications.

Records and reports of in-service valve tests will be kept in accordance with ISTC-9000 of the Code.

3.9(B).7 References

1. WREM-Water Reactor Evaluation Model, Rev. 1, NUREG-75/056, May 1975, NRC, Div. of Technical Review.

2. Public Service Company of New Hampshire letter, dated Jan. 4, 1980, to NRC, Region I, Office of Inspection and Enforcement (response to IE Bulletin No. 79-02, Rev. 2).

3.9(N) MECHANICAL SYSTEMS AND COMPONENTS

3.9(N).1 Special Topics for Mechanical Components

3.9(N).1.1 Design Transients

The following five operating conditions, as defined in Section III of the ASME B&PV Code, are considered in the design of the Reactor Coolant System (RCS) and RCS component supports:

a. Normal Conditions

Any condition in the course of startup, operation in the design power range, hot standby and system shutdown, other than upset, emergency, faulted or testing conditions.

b. Upset Conditions (Incidents of Moderate Frequency)

Any deviations from normal conditions anticipated to occur often enough that the design should include a capability to withstand the conditions without operational impairment. The upset conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power. Upset conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an upset condition shall be included in the design specifications.

c. Emergency Conditions (Infrequent Incidents)

Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. 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. The total number of postulated occurrences for such events shall not cause more than twenty-five stress cycles having an S_a value greater than that for 10^6 cycles from the applicable fatigue design curves of the ASME Code Section III.

d. Faulted Conditions (Limiting Faults)

Those combinations of conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the Nuclear Energy System may be impaired to the extent that considerations of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

e. Testing Conditions

Testing conditions are those pressure overload tests including hydrostatic tests and pneumatic tests. Other types of tests shall be classified under Normal, Upset, Emergency or Faulted Conditions.

To provide the necessary high degree of integrity for the equipment in the RCS, the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in a plant. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses are based upon engineering judgment and experience. The transients selected are representative of operating conditions which prudently should be considered to occur during plant operation, and which are sufficiently severe or frequent to be of possible significance to component cyclic behavior. The transients selected may be regarded as a conservative representation of transients which, used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the plant.

The following design conditions are given in the equipment specifications for RCS components.

The design transients and the number of cycles of each that are normally used for fatigue evaluations are shown in Table 3.9(N)-1. In accordance with ASME III, Emergency and Faulted Conditions are not included in fatigue evaluations.

a. Normal Conditions

The following primary system transients are considered normal conditions:

Heatup and Cooldown at 100°F per hour

Unit Loading and Unloading at 5 Percent of Full Power per Minute

Step Load Increase and Decrease of 10 Percent of Full Power

Large Step Load Decrease with Steam Dump

Steady-State Fluctuations

Initial

Random

Feedwater Cycling at Hot Shutdown

Loop Out of Service

Feedwater Heaters Out of Service

a. One Heater Out of Service

b. One Bank of Heaters Out of Service

Unit Loading and Unloading Between 0 and 15 Percent of Full Power

Boron Concentration Equalization

Refueling

Reduced Temperature Return to Power

Reactor Coolant Pumps Startup and Shutdown

Turbine Roll Test

Primary Side Leak Test

Secondary Side Leak Test

Tube Leakage Test.

1. <u>Heatup and Cooldown at 100°F Per Hour</u>

The design heatup and cooldown cases are conservatively represented by continuous operations performed at a uniform temperature rate of 100°F per hour. (These operations can take place at lower rates approaching the minimum of 0°F per hour.)

For these cases, the heatup occurs from ambient (assumed to be 120°F) to the no-load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hour will not be attained because of other limitations such as:

(a) Material ductility considerations which establish maximum permissible temperature rates of change, as a function of plant pressure and temperature, may be below the design rate of 100°F per hour.

(b) Slower initial heatup rates when using pump energy only.

(c) Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry and gas adjustments.

The number of such complete heatup and cooldown operations is specified as 200 each, which corresponds to five such occurrences per year for the 40-year plant design life.

2. <u>Unit Loading and Unloading at 5 Percent of Full Power per Minute</u>

The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5 percent per minute between 15 percent load and full load. This load swing is the maximum possible consistent with operation under automatic reactor control. The reactor temperature will vary with load as prescribed by the Reactor Control System. The number of loading and unloading operations is defined as 13,200. One loading operation per day yields 14,600 such operations during the 40-year design life of the plant. By assuming a 90 percent availability factor, this number is reduced to 13,200.

3. Step Load Increase and Decrease of 10 Percent of Full Power

The ±10 percent step change in load demand is a transient which is assumed to be a change in turbine control valve opening due to disturbances in the electrical network into which the plant output is tied. The Reactor Control System is designed to restore plant equilibrium without reactor trip following a ±10 percent step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15 percent and 100 percent full load, the power range for automatic reactor control. In effect, during load change conditions, the Reactor Control System attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed setpoint at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step decrease in turbine load, the secondary side steam pressure and_temperature initially increase, since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the RCS average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With the load decrease, the reactor coolant temperature will ultimately be reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature setpoint change is made as a function of turbine-generator load, as determined by first stage turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash, which reduces the rate of pressure decrease. Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and, eventually, the system pressure is restored to its normal value. The reactor coolant average temperature will be raised to a value above its initial equilibrium value at the beginning of the transient.

The number of each operation is specified at 2000 times or 50 per year for the 40-year plant design life.

4. Large Step Load Decrease with Steam Dump

This transient applies to a step decrease in turbine load from full power, of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate a secondary side steam dump that will prevent both reactor trip and lifting of steam generator and pressurizer safety valves. Thus, since this plant is designed to accept a step decrease of 50 percent from full power, the Steam Dump System provides the heat sink to accept the difference in allowable unloading rates between the turbine and the RCS.

The number of occurrences of this transient is specified at 200 times or 5 per year for the 40-year plant design life.

5. Steady-State Fluctuations

It is assumed that the reactor coolant temperature and pressure at any point in the system vary around the nominal (steady-state) values. For design purposes two cases are considered:

(a) Initial Fluctuations - These are due to control rod cycling during the first 20 full power months of reactor operation. Temperature is assumed to vary ±3°F and pressure by ±25 psi, once during each 2 minute period. The total number of occurrences is limited to $1.5x10^5$. These fluctuations are assumed to occur consecutively, and not simultaneously with the random fluctuations.

(b) Random Fluctuations - Temperature is assumed to vary by ±0.5°F and pressure by ±6 psi, once every 6 minutes. With a 6 minute period, the total number of occurrences during the plant design life does not exceed 3.0×10^6.

6. Feedwater Cycling at Hot Shutdown

These transients can occur when the plant is at "no load" conditions, during which intermittent feeding of 32°F feedwater into the steam generators is assumed. Due to fluctuations arising from this mode of operation, the reactor coolant average temperature decreases to a lower value and then immediately begins to return to normal no-load temperature. This transient is assumed to occur 2000 times over the life of the plant.

7. Loop Out of Service

The plant may be operated at a reduced power level with a single loop out of service for limited periods of time. This is accomplished by reducing power level and tripping a single reactor coolant pump.

It is assumed that this transient occurs twice per year or 80 times in the life of the plant. Conservatively, it is assumed that all 80 occurrences can occur in the same loop. In other words, it must be assumed that the whole RCS is subjected to 80 transients, while each loop is also subjected to 80 inactive loop transients.

When an inactive loop is brought back into service, the power level is reduced to approximately 10 percent and the pump is started. It is assumed that an inactive loop is inadvertently started up at maximum allowable power level 10 times over the life of the plant. (This transient is covered under Upset Conditions.) Thus, the normal startup of an inactive loop is assumed to occur 70 times during the life of the plant.

8. Feedwater Heaters Out of Service

These transients occur when one or more feedwater heaters are taken out of service. During the period of time that the heaters are out of service, it is desirable to have the operator maintain the plant at full rated thermal load. To accomplish this, the operator will:

1. Calculate the appropriate steam flow reduction, which will maintain the plant at full rated thermal load after the heater has been taken out of service.

2. Reduce steam flow by the appropriate amount (calculated in 1 above) and allow plant conditions to reach a new steady-state (approximately in 10 minutes.)

3. Take heater (or heaters) out of service.

The number of occurrences of each of these transients is specified at 120 occurrences over the 40-year plant design life.

9. Unit Loading and Unloading Between 0 and 15 Percent of Full Power

The unit loading and unloading cases between 0 and 15 percent power are represented by continuous and uniform ramp power changes, requiring 30 minutes for loading and 5 minutes for unloading. During loading, reactor coolant temperatures are increased from the no-load value to the normal load program temperatures at the 15 percent power level. The reverse temperature change occurs during unloading.

Prior to loading, it is assumed that the plant is at hot shutdown conditions, with 32°F feedwater cycling. During the two-hour period following the beginning of loading, the feedwater temperature increases from 32°F to 300°F due to steam dump and turbine startup heat input to the feedwater. Subsequent to unloading, feedwater heating is terminated, steam dump is reduced to residual heat removal requirements, and feedwater temperature decays from 300°F to 32°F.

The number of these loading and unloading transients is assumed to be 500 each during the 40-year plant design life, which is equivalent to about one occurrence per month.

10. Boron Concentration Equalization

Following any large change in boron concentration in the RCS, spray is initiated in order to equalize concentration between the loops and the pressurizer. This can be done by manually operating the pressurizer backup heater, thus causing a pressure increase, which will initiate spray at a compensated pressurizer pressure of approximately 2275 psia. The proportional sprays return the pressure to 2250 psia, and maintain this pressure by matching the heat input from the backup heater until the concentration is equalized. For design purposes, it is assumed that this operation is performed once after each load change in the design load follow cycle. With two load changes per day, and a 90 percent plant availability factor over the 40-year design life, the total number of occurrences is 26,400.

11. Refueling

At the end of plant cooldown, the temperature of the fluid in the RCS is less than 125°F. At this time, the vessel head is removed and the refueling canal is filled. This is done by pumping water from the refueling water storage tank, which is outside and conservatively assumed to be at 32°F, into the loops by means of the low head safety injection pumps. The refueling water flows directly into the reactor vessel via the accumulator connections and cold legs.

This operation is assumed to occur twice per year, or 80 times over the life of the plant.

12. Reduced Temperature Return to Power

The reduced temperature return to power operation is designed to improve the spinning reserve capabilities of the plant during load-follow operations without part length rods. The transient will normally begin at the ebb (50 percent) of a load-follow cycle and will proceed at a rapid positive rate (typically 5 percent per minute) until the abilities of the control rods and the coolant temperature reduction (negative moderator coefficient) to supply reactivity are exhausted. At that point, further power increases are limited to approximately one percent per minute by the ability of the boron system to dilute the reactor coolant. The reduction in primary coolant temperature is limited by the protection system to about 20°F below the programmed value.

The reduced temperature return to power operation is not intended for daily use. It is designed to supply additional plant capabilities when required because of network fault or upset condition. Hence this mode of operation is not expected to be used more than once a week in practice (2000 times in 40 years).

13. <u>Reactor Coolant Pumps (RCP) Startup and Shutdown</u>

The reactor coolant pumps are started and stopped during routine operations such as RCS venting, plant heatup and cooldown, and in connection with recovery from certain transients such as Loop Out of Service and Loss of Power. Other (undefined) circumstances may also require pump starting and stopping.

Of the spectrum of RCS pressure and temperature conditions under which these operations may occur, three conditions have been selected for defining transients:

Cold condition - 70°F and 400 psig

Pump restart condition - 100°F and 400 psig

Hot condition - 557°F and 2235 psig

For RCP starting and stopping operations, it is assumed that variations in RCS primary side temperature and in pressurizer pressure and temperature are negligible, and that the steam generator secondary side is completely unaffected. The only significant variables are the primary system flow and the pressure changes resulting from the pump operations.

The following cases were considered:

<u>Case 1 - First Pump Startup (Last Pump Shutdown)</u>

Variations in reactor coolant loop flow accompany startup of the first pump, both in the loop containing the pump being started and in the other loops (loops in which the pumps remain idle). This case involves a higher dynamic pressure loss in the loop containing the pump being started, but the magnitude of the flow change is less than in Case 2. For the last pump shutdown case, the transient is the reverse of the first pump startup transient.

Case 2 - Last Pump Startup (First Pump Shutdown)

This case conservatively represents the variations in reactor coolant loop flow accompanying startup of the second and third pumps. Initially, flow exists through these loops in the reverse direction as the result of starting the first pumps. The remaining pumps are then started in sequence, and a new equilibrium flow is established. The magnitude of flow reversal is the largest in the loop containing the last pump to be started. For the first pump shutdown case, the transient is the reverse of the last pump startup transient.

Design values for the pump starting/stopping conditions are given in Table 3.9(N)-1, along with the assumed number of occurrences. The 3800 occurrences listed in the table do not include the startups or shutdowns associated with RCS heatup and cooldown.

14. Turbine Roll Test

This transient is imposed upon the plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power will be used to heat the reactor coolant to operating temperature (no-load conditions), and the steam generated will be used to perform a turbine roll test. However, the plant cooldown during this test will exceed the 100°F per hour design rate.

The number of such test cycles is specified at 20 times, to be performed at the beginning of plant operating life prior to irradiation.

15. Primary Side Leakage Test

Subsequent to each time the primary system has been opened, a leakage test will be performed. During this test, the primary system pressure is, for design purposes, raised to 2500 psia, with the system temperature above the minimum temperature imposed by reactor vessel material ductility requirements, while the system is checked for leaks.

In actual practice, the primary system will be pressurized to approximately 2400 psig, as measured at the pressurizer, to prevent the pressurizer safety valves from lifting during the leak test.

During this leakage test, the secondary side of the steam generator must be pressurized so that the pressure differential across the tube sheet does not exceed 1600 psi. This is accomplished with the steam, feedwater, and blowdown lines closed off. For design purposes, it is assumed that 200 cycles of this test will occur during the 40-year life of the plant.

16. Secondary Side Leakage Test

During the life of the plant, it may be necessary to check the secondary side of the steam generator (particularly, the manway closure) for leakage. For design purposes, it is assumed that the steam generator secondary side is pressurized to just below its design pressure, to prevent the safety valves from lifting. In order not to exceed a secondary side to primary side pressure differential of 670 psi, the primary side must also be pressurized. The primary system must be above the minimum temperature imposed by reactor vessel materials ductility requirements. It is assumed, that this test is performed 80 times during the 40-year life of the plant.

17. Tube Leakage Test

During the life of the plant, it may be necessary to check the steam generator for tube leakage and tube to tube sheet leakage. This is done by visual inspection of the underside (channel head side) of the tube sheet for water leakage, with the secondary side pressurized. Tube leakage tests are performed during plant cold shutdowns.

For these tests, the secondary side of the steam generator is pressurized with water, initially at a relatively low pressure, and the primary system remains depressurized. The underside of the tube sheet is examined visually for leaks. If any are observed, the secondary side is then depressurized and repairs made by tube plugging. The secondary side is then repressurized (to a higher pressure), and the underside of the tube sheet is again checked for leaks. This process is repeated until all the leaks are repaired. The maximum (final) secondary side test pressure reached is 840 psig.

The total number of tube leakage test cycles is defined as 800 during the 40-year life of the plant. Following is a breakdown of the anticipated number of occurrences at each secondary side test pressure:

Test Pressure, psig	Number Occurrences
200	400
400	200
600	120
840	80

Both the primary and secondary sides of the steam generators will be at the ambient temperatures during these tests.

b. Upset Conditions

The following primary system transients are considered upset conditions:

Loss of Load (without immediate reactor trip)

Loss of Power

Partial Loss of Flow

Reactor Trip from Full Power

Inadvertent Reactor Coolant System Depressurization

Inadvertent Startup of an Inactive Loop

Control Rod Drop

Inadvertent Safety Injection Actuation

Operating Basis Earthquake

Excessive Feedwater Flow

RCS Cold Overpressurization

1. Loss of Load (Without Immediate Reactor Trip)

   This transient applies to a step decrease in turbine load from full power (turbine trip) without immediately initiating a reactor trip, and represents the most severe pressure transient on the RCS under upset conditions. The reactor eventually trips as a consequence of a high pressurizer level trip initiated by the Reactor Protection System (RPS). Since redundant means of tripping the reactor are provided as a part of the RPS, transients of this nature are not expected, but are included to ensure a conservative design.

   The number of occurrences of this transient is specified at 80 times, or 2 times per year for the 40-year plant design life.

2. Loss of Power

   This transient applies to a blackout situation involving the loss of outside electrical power to the station, assumed to be operating initially at 100 percent power, followed by reactor and turbine trips. Under these circumstances, the reactor coolant pumps are de-energized and, following coastdown of the reactor coolant pumps, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the steam generators which, at this time, are receiving feedwater, assumed to be at 32°F, from the Emergency Feedwater System operating from diesel generator power. Steam is removed for reactor cooldown through atmospheric relief valves provided for this purpose.

   The number of occurrences of this transient is specified at 40 times or 1 per year for the 40-year plant design life.

3. Partial Loss of Flow

   This transient applies to a partial loss of flow from full power, in which a reactor coolant pump is tripped out of service as the result of a loss of power to that pump. The consequences of such an accident are a reactor and turbine trip, on low reactor coolant flow, followed by automatic opening of the Steam Dump System and flow reversal in the affected loop. The flow reversal causes reactor coolant at cold leg temperature to pass through the steam generator and be cooled still further. This cooler water then flows through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizable reduction in the hot leg coolant temperature of the affected loop.

The number of occurrences of this transient is specified at 80 times, or 2 times per year for the 40-year plant design life.

4. <u>Reactor Trip From Full Power</u>

A reactor trip from full power may occur from a variety of causes resulting in temperature and pressure transients in the RCS and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the RPS causes the control rods to move into the core.

Various moderator cooldown transients associated with reactor trips can occur as a result of excessive feed or steam dump after trip or large load increase. For design purposes, reactor trip is assumed to occur a total of 400 times or 10 times per year over the life of the plant. The various types of trips and the number of occurrences for each are as follows:

(a) Reactor trip with no inadvertent cooldown – 230 occurrences

(b) Reactor trip with cooldown but no safety injection - 160 occurrences

(c) Reactor trip with cooldown actuating safety injection – 10 occurrences.

5. <u>Inadvertent Reactor Coolant System Depressurization</u>

Several events can be postulated as occurring during normal plant operation which will cause rapid depressurization of the RCS. These include:

(a) Actuation of a single pressurizer safety valve

(b) Inadvertent opening of one pressurizer power-operated relief valve due either to equipment malfunction or operator error

(c) Malfunction of a single pressurizer pressure controller causing one power-operated relief valve and two pressurizer spray valves to open

(d) Inadvertent opening of one pressurizer spray valve, due either to equipment malfunction or operator error

(e) Inadvertent auxiliary spray.

Of these events, the pressurizer safety valve actuation causes the most severe transient, and is used as an "umbrella" case to conservatively represent the reactor coolant pressure and temperature variations arising from any of them.

When a pressurizer safety valve opens, and remains open, the system rapidly depressurizes, the reactor trips, and the Safety Injection System (SIS) is actuated. Also, the passive accumulators of the SIS are actuated when pressure decreases by approximately 1600 psi, about 12 minutes after the depressurization begins. The depressurization and cooldown are eventually terminated by operator action. All of these effects are completed within approximately 18 minutes. It is conservatively assumed that none of the pressurizer heaters are energized.

With pressure constant and safety injection in operation, boiloff of hot leg liquid through the pressurizer and open safety valve will continue.

For design purposes, this transient is assumed to occur 20 times during the 40-year design life of the plant.

6. Inadvertent Startup of an Inactive Loop

This transient can occur when a loop is out of service. With the plant operating at maximum allowable power level, the reactor coolant pump in the inactive loop is started as a result of operator error. Reactor trip occurs on high nuclear flux. This transient is assumed to occur 10 times during the life of the plant.

7. Control Rod Drop

This transient occurs if a bank of control rods drops into the fully inserted position due to a single component failure. It is assumed that this transient occurs 80 times over the life of the plant.

8. <u>Inadvertent Safety Injection Actuation</u>

A spurious safety injection signal results in an immediate reactor trip followed by actuation of the high head centrifugal charging pumps. These pumps deliver borated water from the RWST to the RCS cold legs. The initial portion of this transient is similar to the Reactor Trip from Full Power with no cooldown. Controlled steam dump and feedwater flow after trip removes core residual heat. Reactor coolant temperature and pressure decrease as the control rods move into the core.

Later in the transient, the injected water causes the RCS pressure to increase to the pressurizer power-operated relief valve setpoint and the primary and secondary temperatures to decrease gradually. The transient continues until the operator stops the charging pumps. It is assumed that the plant is then returned to no-load conditions, with pressure and temperature changes controlled within normal limits.

For design purposes, this transient is assumed to occur 60 times during the 40-year design life of the plant.

9. <u>Operating Basis Earthquake</u>

The mechanical stresses resulting from the Operating Basis Earthquake are considered on a component basis. Fatigue analysis, where required by the codes, is performed by the supplier as part of the stress analysis report. The earthquake loads are a part of the mechanical loading conditions specified in the equipment specifications. The origin of their determination is separate and distinct from those transients resulting from fluid pressure and temperature. They are, however, considered in the design analysis.

10. <u>Excessive Feedwater Flow</u>

An excessive feedwater flow transient is conservatively defined as an "umbrella" case to cover occurrence of several events of the same general nature. The postulated transient results from inadvertent opening of a feedwater control valve while the plant is at the hot standby or no load condition, with the feedwater, condensate and heater drain systems in operation.

It is assumed, that the stem of a feedwater control valve fails and the valve immediately reaches the full open position. In the steam generator directly affected by the malfunctioning valve ("failed loop"), the feedwater flow step increases from essentially zero flow to the value determined by the system resistance and the developed head of all operating feedwater pumps. Steam flow is assumed to remain at zero, and the temperature of the feedwater entering the steam generator is conservatively assumed to be 32°F. Feedwater flow is isolated on a reactor coolant low T_{avg} signal; a low pressurizer pressure signal actuates the Safety Injection System. Emergency feedwater flow, initiated by the safety injection signal, is assumed to continue with all pumps discharging into the affected steam generator. It is assumed, for conservatism in the secondary side analysis, that there is emergency feedwater flow to the steam generators not affected by the malfunctioned valve in the "unfailed loops." Plant conditions stabilize at the values reached in 600 seconds, at which time emergency feedwater flow is terminated. The plant is then either taken to cold shutdown, or returned to the no load condition at a normal heatup rate with the Emergency Feedwater System under manual control.

For design purposes, this transient is assumed to occur 30 times during the life of the plant.

11. <u>RCS Cold Overpressurization</u>

RCS cold overpressurization occurs during startup and shutdown conditions at low temperature, with or without the existence of a steam bubble in the pressurizer, and is especially severe when the reactor coolant system is in a water-solid configuration. The event is inadvertent, and can potentially occur by any one of a variety of malfunctions or operator errors. All events, which have occurred to date, may be categorized as belonging to either events resulting in the addition of mass (mass input transients) or events resulting in the addition of heat (heat input transients). All these possible transients are represented by composite, “umbrella” design transients, referred to here as RCS cold overpressurization.

The number of occurrences of this transient is specified at 10 times for the 40-year plant design operating period.

c. Emergency Conditions

The following primary system transients are considered Emergency Conditions:

Small Loss-of-Coolant Accident

Small Steam Line Break

Complete Loss of Flow.

1. Small Loss-of-Coolant Accident

For design transient purposes, the small loss-of-coolant accident is defined as a break equivalent to the severance of a 1-inch inside diameter branch connection. (Breaks smaller than 0.375 inch inside diameter can be handled by the normal makeup system, and produce no significant fluid systems transients.) Breaks which are much larger than 1 inch will cause safety injection soon after the accident, and are regarded as Faulted Conditions. For design purposes, it is assumed that this transient occurs five times during the life of the plant. It is assumed that the Safety Injection System is actuated immediately after the break occurs, and subsequently delivers water at a minimum temperature of 32°F to the RCS.

2. Small Steam Line Break

For design transient purposes, a small steam line break is defined as a break equivalent in effect to a steam safety valve opening and remaining open. This transient is assumed to occur five times during the life of the plant. The following conservative assumptions are used in defining the transients:

(a) The reactor is initially in a hot, zero-power condition.

(b) The small steam line break results in immediate reactor trip and safety injection actuation.

(c) A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.

(d) The Safety Injection System operates at design capacity and repressurizes the RCS within a relatively short time.

3. Complete Loss of Flow

   This accident involves a complete loss of flow from full power, resulting from simultaneous loss of power to all reactor coolant pumps. The consequences of this incident are a reactor trip and turbine trip on undervoltage, followed by automatic opening of the Steam Dump System. For design purposes, this transient is assumed to occur five times during the plant lifetime.

d. Faulted Conditions

The following primary system transients are considered Faulted Conditions. Each of the following accidents should be evaluated for one occurrence:

Reactor Coolant Pipe Break (Large Loss-of-Coolant Accident)

Large Steam Line Break

Feedwater Line Break

Reactor Coolant Pump Locked Rotor

Control Rod Ejection

Steam Generator Tube Rupture

Safe Shutdown Earthquake.

1. Reactor Coolant Pipe Break (Large Loss-of-Coolant Accident)

   Following rupture of a reactor coolant pipe resulting in a large loss of coolant, the primary system pressure decreases causing the primary system temperature to decrease. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal will still be at or near the operating temperature by the end of blowdown. It is conservatively assumed that the Safety Injection System is actuated to introduce water at a minimum temperature of 32°F into the RCS. The safety injection signal will also result in reactor and turbine trips.

2. Large Steam Line Break

This transient is based on the complete severance of the largest steam line. The following conservative assumptions were made:

(a) The reactor is initially in a hot, zero-power condition.

(b) The large steam line break results in immediate reactor trip and in actuation of the Safety Injection Systems.

(c) A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transients.

(d) The Safety Injection System operates at design capacity and repressurizes the Reactor Coolant System within a relatively short time.

3. Feedwater Line Break

This accident involves a double-ended rupture of the main feedwater piping from full power, resulting in the rapid blowdown of one steam generator and the termination of main feedwater flow to the others. The blowdown is completed in approximately 27 seconds. Conditions were conservatively chosen to give the most severe primary side and secondary side transients. All emergency feedwater flow exits at the break. The incident is terminated when the operator manually realigns the Emergency Feedwater System to isolate the break and to deliver emergency feedwater to the intact steam generators.

4. Reactor Coolant Pump Locked Rotor

This accident is based on the instantaneous seizure of a reactor coolant pump with the plant operating at full power. The locked rotor can occur in any loop. Reactor trip occurs almost immediately, as the result of low coolant flow in the affected loop.

5. Control Rod Ejection

This accident is based on the single most reactive control rod being instantaneously ejected from the core. This reactivity insertion in a particular region of the core causes a severe pressure increase in the RCS such that the pressurizer safety valves will lift and also causes a more severe temperature transient in the loop associated with the affected region than in the other loops. For conservatism the analysis is based on the reactivity insertion and does not include the mitigating effects (on the pressure transient) of coolant blowdown through the hole in the vessel head vacated by the ejected rod.

6. Steam Generator Tube Rupture

This accident postulates the double-ended rupture of a steam generator tube resulting in a decrease in pressurizer level and reactor coolant pressure. Reactor trip will occur due to the resulting safety injection signal. In addition, safety injection actuation automatically isolates the feedwater lines, by tripping all feedwater pumps and closing the feedwater isolation valves. When this accident occurs, some of the reactor coolant blows down into the affected steam generator causing the shell side level to rise. The primary system pressure is reduced below the secondary safety valve setting. Subsequent recovery procedures call for isolation of the steam line leading from the affected steam generator. This accident will result in a transient which is no more severe than that associated with a reactor trip from full power. Therefore, it requires no special treatment insofar as fatigue evaluation is concerned, and no specific number of occurrences is postulated.

7. Safe Shutdown Earthquake

The mechanical dynamic or static equivalent loads due to the vibratory motion of the Safety Shutdown Earthquake are considered on a component basis.

e. Test Conditions

The following primary system transients under Test Conditions are discussed:

Primary Side Hydrostatic Test

Secondary Side Hydrostatic Test.

1. Primary Side Hydrostatic Test

   The pressure tests include both shop and field hydrostatic tests which occur as a result of component or system testing. This hydro test is performed at a water temperature which is compatible with reactor vessel material ductility requirements and a test pressure of 3107 psig (1.25 times design pressure). In this test, the RCS is pressurized to 3107 psig, coincident with steam generator secondary side pressure of 0 psig. The RCS is designed for 10 cycles of these hydrostatic tests, which are performed prior to plant startup. The number of cycles is independent of other operating transients.

   Additional hydrostatic tests will be performed to meet the in-service inspection requirements of ASME Section XI, Subarticle IS5-20. A total of four such tests is expected. The increase in the fatigue usage factor caused by these tests is easily covered by the conservative number (200) of primary side leakage tests that are considered for design.

2. Secondary Side Hydrostatic Test

   The secondary side of the steam generator is pressurized to 1.25 design pressure with a minimum water temperature of 120°F, coincident with the primary side at 0 psig.

   For design purposes, it is assumed that the steam generator will experience 10 cycles of this test.

   These tests may be performed either prior to plant startup, or subsequently following shutdown for major repairs or both.

3.9(N).1.2 Computer Programs Used in Analyses

The following computer programs have been used in dynamic and static analyses to determine mechanical loads, stresses, and deformations of seismic Category I components and equipment. These are described and verified in References 1 and 2.

a. WESTDYN

   Static and dynamic analysis of redundant piping systems

b. FIXFM-3

Time-history response of three-dimensional structures

c. WESDYN-2

Piping system stress analysis from time-history displacement data

d. STHRUST

Hydraulic loads on loop components from blowdown information

e. WESAN

Reactor coolant loop equipment support structures analysis and evaluation

f. WECAN

Finite element structural analysis

g. DARIWOSTAS

Dynamic transient response analysis of reactor vessel and internals.

3.9(N).1.3 Experimental Stress Analysis

No experimental stress analysis methods are used for seismic Category I systems or components. However, Westinghouse makes extensive use of measured results from prototype plants and various scale model tests, as discussed in Subsection 3.9(N).2.

3.9(N).1.4 Considerations for the Evaluation of the Faulted Condition

a. Loading Conditions

The structural stress analyses performed on the Reactor Coolant System consider the loadings specified as shown in Table 3.9(N)-2. These loads result from thermal expansion, pressure, weight, Operating Basis Earthquake (OBE), Safe Shutdown Earthquake (SSE), design basis loss-of-coolant accident, and plant operational thermal and pressure transients.

b. Analysis of the Reactor Coolant Loop

The reactor coolant loop piping is evaluated in accordance with the Criteria of ASME III, NB-3650 and Appendix F. The loads included in the evaluation result from the SSE, dead weight, pressure, and LOCA loadings from the effect of the three branch nozzle breaks per Subsection 3.6(N).2.1 (loop hydraulic forces, asymmetric sub-compartment pressurization forces, and reactor vessel motion) and the secondary side breaks at the main steam line and main feedwater line terminal end nozzle locations at the steam generator.

The loads used in the analysis of the Reactor Coolant Loop/Support System are described in detail below.

1. Pressure

Pressure loading is identified as either membrane design pressure or general operating pressure, depending upon its application. The membrane design pressure is used in connection with the longitudinal pressure stress and minimum wall thickness calculations in accordance with the ASME Code.

The term operating pressure is used in connection with determination of the system deflections and support forces. The steady-state operating hydraulic forces based on the system initial pressure are applied as general operating pressure loads to the reactor coolant loop model at changes in direction or flow area.

2. Weight

A weight analysis is performed to meet Code requirements by applying a 1.0 g load downward on the complete piping system. The piping is assigned a distributed mass or weight as a function of its properties. This method provides a distributed loading to the piping system as a function of the weight of the pipe and contained fluid during normal operating conditions.

3. Seismic

The forcing functions for the reactor coolant loop piping seismic analyses are three orthogonal components of earthquake. The three components used to simulate the earthquake are in the form of statistically independent time-history accelerations. The earthquake accelerations for the north-south direction and east-west direction are applied to the containment base mat in the respective global directions simultaneously with the vertical acceleration along the vertical direction.

For the OBE and SSE seismic analyses, damping values as defined in Section 3.7(N) are used in the reactor coolant loop/supports system analysis.

4. Loss-of-Coolant Accident

Blowdown loads are developed in the reactor coolant loop as a result of transient flow and pressure fluctuations following a postulated pipe break in one of the three large RCL branch nozzles. Structural consideration of dynamic effects of postulated pipe breaks requires postulation of a finite number of break locations. Postulated pipe break locations are given in Section 3.6(N).

Time-history dynamic analysis is performed for these postulated break cases. Hydraulic models are used to generate time-dependent hydraulic forcing functions used in the analysis of the reactor coolant loop for each break case. For a further description of the hydraulic forcing functions, refer to Section 3.6(N).

5. Transients

The Code requires satisfaction of certain requirements relative to operating transient conditions. Operating transients are discussed in Subsection 3.9(N).1.1.

The vertical thermal growth of the reactor pressure vessel nozzle centerlines is considered in the thermal analysis to account for equipment nozzle displacement as an external movement.

The hot moduli of elasticity E, the coefficient of thermal expansion at the metal temperature ∝, the external movements transmitted to the piping due to vessel growth, and the temperature rise above the ambient temperature ΔT, define the required input data to perform the flexibility analysis for thermal expansion.

To provide the necessary high degree of integrity for the Reactor Coolant System, the transient conditions selected for fatigue evaluation are based on conservative estimates of the magnitude and anticipated frequency of occurrence of the temperature and pressure transients resulting from various plant operating conditions.

c. Reactor Coolant Loop Analytical Models and Methods

The analytical methods used in obtaining the solution consist of the transfer matrix method and stiffness matrix formulation for the static structural analysis, time-history integration method for seismic dynamic analysis, and time-history integration method for the loss-of-coolant accident dynamic analysis.

The integrated reactor coolant loop/supports system model is the basic system model used to compute loadings on components, component supports, and piping. The system model includes the stiffness and mass characteristics of the reactor coolant loop piping and components, the stiffness of supports, and the stiffnesses of auxiliary line piping which affect the system. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

1. Static

The reactor coolant loop/supports system model, constructed for the WESTDYN computer program, is represented by an ordered set of data which numerically describes the physical system. Figure 3.9(N)-1 shows an isometric line schematic of this mathematical model. The steam generator and reactor coolant pump vertical and lateral support members are described in Subsection 5.4.14.

The spatial geometric description of the reactor coolant loop model is based upon the reactor coolant loop piping layout and equipment drawings. The node point coordinates and incremental lengths of the members are determined from these drawings. Geometrical properties of the piping and elbows along with the modulus of elasticity E, the coefficient of thermal expansion $\propto$, the average temperature change from ambient temperature ΔT, and the weight per unit length are specified for each element. The primary equipment supports are represented by stiffness matrices which define restraint characteristics of the supports. Due to the symmetry of the static loadings, the reactor pressure vessel centerline is represented by a fixed boundary in the system mathematical model. The vertical thermal growth of the reactor vessel nozzle centerline is considered in the construction of the model.

The model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained. These loads are incorporated into the overall load vector.

After all the sections have been defined in this matter, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system.

The static solutions for weight, thermal, and general pressure loading conditions are obtained by using the WESTDYN computer program. The derivation of the hydraulic loads for the loss-of-coolant accident analysis of the loop is covered in Subsection 3.6(N).2.

2. Seismic

The model used in the static analysis is modified for the dynamic analysis by including the mass characteristics of the piping and equipment. In the time-history seismic analysis, the containment internals structure and all of the piping loops are included in the coupled system model. The effect of the equipment motion on the Reactor Coolant Loop/Supports System is obtained by modeling the mass and the stiffness characteristics of the equipment in the overall system model.

The steam generator is represented by three discrete masses. The lowest mass is located at the intersection of the centerlines of the inlet and outlet nozzles of the steam generator. The middle mass is located at the steam generator upper support elevation and the third mass is located at the top of the steam generator.

The reactor coolant pump is typically represented by a two discrete mass model. The lower mass is located at the intersection of the centerlines of the pump suction and discharge nozzles. The upper mass is located near the center of gravity of the motor.

The reactor vessel and core internals are represented by three discrete masses. The masses are lumped at various locations representing the pressure vessel and reactor internals.

The component upper and lower lateral supports are inactive during plant heatup, cooldown and normal plant operating conditions. However, these restraints become active when the plant is at power and under the rapid motions of the reactor coolant loop components that occur from the dynamic loadings and are represented by stiffness matrices and/or individual tension or compression spring members in the dynamic model. The analyses are performed at the full power condition.

The total response is obtained directly by direct time integration of the equations of motion. The results of the time-history analysis are forces and displacements. The time-history displacement response is then used in computing support loads and in performing the reactor coolant loop piping stress evaluation.

3. Loss-of-Coolant Accident

The mathematical model used in the static analyses is modified for the loss-of-coolant accident analyses by including the mass characteristics of the piping and primary equipment. The natural frequencies and eigenvectors are determined from this.

The time-history hydraulic forces at the node points are combined to obtain the forces and moments acting at the corresponding structural lumped-mass node points.

The dynamic structural solution for the full power loss-of-coolant accident and steam line break is obtained by using a modified-predictor-corrector-integration technique and normal mode theory.

When elements of the system can be represented as single acting members (tension or compression members), they are considered as nonlinear elements, which are represented mathematically by the combination of a gap, a spring, and a viscous damper. The force in this nonlinear element is treated as an externally applied force in the overall normal mode solution. Multiple nonlinear elements can be applied at the same node, if necessary.

The time-history solution is performed in program FIXFM3. The input to this subprogram consists of the natural frequencies, normal modes, applied forces and nonlinear elements. The natural frequencies and normal modes for the modified reactor coolant loop dynamic model are determined with the WESTDYN program. To properly simulate the release of the strain energy in the pipe, the internal forces in the system at the postulated break location due to the initial steady-state hydraulic forces, thermal forces, and weight forces are determined. The release of the strain energy is accounted for by applying the negative of these internal forces as a step function loading. The initial conditions are equal to zero because the solution is only for the transient problem (the dynamic response of the system from the static equilibrium position). The time-history displacement solution of all dynamic degrees of freedom is obtained using program FIXFM3 and employing 4 percent critical damping.

The loss-of-coolant accident displacements of the reactor vessel are applied in time-history form as input to the dynamic analysis of the reactor coolant loop. The loss-of-coolant accident analysis of the reactor vessel includes all the forces acting on the vessel including internals reactions and loop mechanical loads. The reactor vessel analysis is described in Subsection 3.9(N).1.5.

The resultant asymmetric external pressure loads on the RCP and steam generator resulting from a postulated pipe rupture and pressure build-up in the loop compartments, are applied to the same integrated RCL/supports system model used to compute loadings on components, component supports and RCL piping as discussed above. The response of the entire system is obtained for the various external pressure loading cases from which the internal member forces and piping stresses are calculated for each pipe break case considered. The equipment support loads and piping stresses resulting from the external pressure loading are added to the support loads and piping stresses calculated using the loop LOCA hydraulic forces and RPV motion.

The break locations considered for subcompartment pressurization are those postulated for the RCL LOCA analysis, as discussed in Section 3.6(N) and WCAP-8172 (Reference 1 of Section 3.6(N)). The asymmetric subcompartment pressure loads are provided to Westinghouse by United Engineers & Constructors Inc. The analysis to determine these loads is discussed in Section 6.2.

The time-history displacement response of the loop is used in computing support loads and in performing stress evaluation of the reactor coolant loop piping.

The time-history displacements of the FIXFM3 (or WESTDYN) program are used as input to WESDYN-2 (or WESTDYN) to determine the internal forces, deflections, and stresses at each end of the piping elements. For this calculation the displacements are treated as imposed deflections on the reactor coolant loop masses.

4. Transients

Operating transients in a nuclear power plant cause thermal and/or pressure fluctuations in the reactor coolant fluid. The thermal transients cause time-varying temperature distributions across the pipe wall. These temperature distributions resulting in pipe wall stresses may be further subdivided in accordance with the Code into three parts, a uniform, a linear, and a nonlinear portion. The uniform portion results in general expansion loads. The linear portion causes a bending moment across the wall and the nonlinear portion causes a skin stress.

The transients as defined in Subsection 3.9(N).1.1 are used to define the fluctuations in plant parameters. A one-dimensional finite difference heat conduction program is used to solve the thermal transient problem.

The pipe is represented by at least fifty elements through the thickness of the pipe. The convective heat transfer coefficient employed in this program represents the time varying heat transfer due to free and forced convection. The outer surface is assumed to be adiabatic while the inner surface boundary experiences the temperature of the coolant fluid. Fluctuations in the temperature of the coolant fluid produce a temperature distribution through the pipe wall thickness which varies with time. An arbitrary temperature distribution across the wall is shown in Figure 3.9(N)-2.

The average through-wall temperature, T_A, is calculated by integrating the temperature distribution across the wall. This integration is performed for all time steps so that T_A is determined as a function of time.

$$T_A(t) = \frac{1}{H} \int_O^H T(X,t)\, dX$$

The range of temperature between the largest and smallest value of T_A is used in the flexibility analysis to generate the moment loadings caused by the associated temperature changes.

The thermal moment about the mid-thickness of the wall caused by the temperature distribution through the wall is equal to:

$$M = E \propto \int_O^H \left(X - \frac{H}{2} \right) T(X,t)\, dX$$

The equivalent thermal moment produced by the linear thermal gradient as shown in Figure 3.9(N)-2 about the mid-wall thickness is equal to:

$$M_L = E \propto \frac{\Delta T_1}{12} H^2$$

Equating M_L and M, the solution for ΔT_1 as a function of time is:

$$\Delta T_1(t) = \frac{12}{H^2} \int_O^H \left(X - \frac{H}{2} \right) T(X, t)\, dX$$

The maximum nonlinear thermal gradient, ΔT_2, will occur on the inside surface and can be determined as the difference between the actual metal temperature on this surface and half of the average linear thermal gradient plus the average temperature.

$$T_{2I}(t) = T(O, t) - T_A(t) - \frac{|\Delta T_1|}{2}$$

5. <u>Load Set Generation</u>

A load set is defined as a set of pressure loads, moment loads, and through-wall thermal effects at a given location and time in each transient. The method of load set generation is based on Reference 3. The through-wall thermal effects are functions of time and can be subdivided into four parts:

(a) Average temperature (T_A) is the average temperature through-wall of the pipe which contributes to general expansion loads.

(b) Radial linear thermal gradient which contributes to the through-wall bending moment (ΔT_1).

(c) Radial nonlinear thermal gradient (ΔT_2) which contributes to a peak stress associated with shearing of the surface.

(d) Discontinuity temperature (T_A - T_B) represents the difference in average temperature at the cross sections on each side of a discontinuity.

Each transient is described by at least two load sets representing the maximum and minimum stress state during each transient. The construction of the load sets is accomplished by combining the following to yield the maximum (minimum) stress state during each transient:

(a) ΔT_1

(b) ΔT_2

(c) $\alpha_A T_A - \alpha_B T_B$

(d) Moment loads due to T_A

(e) Pressure loads.

This procedure produces at least twice as many load sets as transients for each point.

For all possible load set combinations, the primary-plus-secondary and peak stress intensities, fatigue reduction factors and cumulative usage factors are calculated. The WESTDYN program is used to perform this analysis in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB-3650. Since it is impossible to predict the order or occurrence of the transients over a forty-year life, it is assumed that the transients can occur in any sequence. This is a very conservative assumption.

The combination of load sets yielding the highest alternating stress intensity range is used to calculate the incremental usage factor. The next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

d. Primary Component Supports Models and Methods

The static and dynamic structural analyses employ the matrix method and normal mode theory for the solution of lumped-parameter, multimass structural models. The equipment support structure models are dual-purpose since they are required to: (1) quantitatively represent the elastic restraints which the supports impose upon the loop, and (2) evaluate the individual support member stresses due to the forces imposed upon the supports by the loop.

A description of the supports is found in Subsection 5.4.14. Detailed models are developed using beam elements and plate elements, where applicable. The reactor vessel supports are modeled using the WECAN computer program. Structure geometry, topology and member properties are used in modeling. Steam generator and reactor coolant pump supports are modeled as linear or nonlinear springs.

For each operating condition, the loads (obtained from the RCL analysis) acting on the support structures are appropriately combined. Reactor coolant loop normal and upset conditions thermal expansion loads are treated as primary loading for the primary component supports. The adequacy of each member of the steam generator supports, reactor coolant pump supports, and pressurizer supports is verified by solving the ASME III Subsection NF stress and interaction equations. The adequacy of the RPV Support Structure is verified using the WECAN computer program and comparing the resultant stresses to the criteria given in ASME III Subsection NF.

The respective computer programs are used with these models to obtain support stiffness matrices and member influence coefficients for the steam generator, reactor coolant pump, pressurizer and reactor vessel supports. Unit force along and unit moment about each coordinate axis are applied to the models at the equipment vertical centerline joint. Stiffness analyses are performed for each unit load for each model.

Joint displacements for applied unit loads are formulated into flexibility matrices. These are inverted to obtain support stiffness matrices which were included in the reactor coolant loop model.

Loads acting on the supports obtained from the reactor coolant loop analysis, support structure member properties, and influence coefficients at each end of each member are input into the WESAN program.

For each support case used, the following is performed:

1. Combine the various types of support plane loads to obtain operating condition loads (Normal, Upset, Emergency or Faulted).

2. Multiply member influence coefficients by operating condition loads to obtain all member internal forces and moments.

3. Solve appropriate stress or interaction equations for the specified operating condition. Maximum normal stress, shear stress, and combined load interaction equation values are printed as a ratio of maximum actual values divided by limiting values. ASME Boiler and Pressure Vessel Code Section III, Subsection NF, stress and interaction equations are used with limits for the operating condition specified.

e. Analysis of Primary Components

Equipment which serves as part of the pressure boundary in the reactor coolant loop includes the steam generators, the reactor coolant pumps, the pressurizer, and the reactor vessel. This equipment is ANS Safety Class 1 and the pressure boundary meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. This equipment is evaluated for the loading combinations outlined in Table 3.9(N)-2. The equipment is analyzed for: (1) the normal loads of deadweight, pressure and thermal, (2) mechanical transients of OBE, SSE, and pipe ruptures, and (3) pressure and temperature transients outlined in Subsection 3.9(N).1.1.

The results of the reactor coolant loop analysis are used to determine the loads acting on the equipment nozzles and the support/component interface locations. These loads are supplied for all loading conditions on an "umbrella" load basis. That is, on the basis of previous plant analysis, a set of loads is determined which should be larger than those seen in any single plant analysis. The "umbrella" loads represent a conservative means of allowing detailed component analysis prior to the completion of the system analysis. Upon completion of the system analysis, conformance is demonstrated between the actual plant loads and the loads used in the analyses of the components. Any deviations where the actual load is larger than the "umbrella" load will be handled by individualized analysis.

Seismic analyses are performed individually for the reactor coolant pump, the pressurizer, and the steam generator. Detailed and complex dynamic models are used for the dynamic analyses. The response spectra corresponding to the building elevation at the highest component/building attachment elevation is used for the component analysis. Seismic analyses for the steam generator and pressurizer are performed using 2 percent damping for the OBE and 4 percent damping for the SSE. The analysis of the reactor coolant pump for determination of loads on the motor, main flange, and pump internals is performed using the damping for bolted steel structures, that is, 4 percent for the OBE and 7 percent for the SSE (2 percent for OBE and 4 percent for SSE is used in the system analysis). This damping is applicable to the reactor coolant pump since the main flange, motor stand, and motor are all bolted assemblies (see Section 5.4). The reactor pressure vessel is qualified by static stress analysis based on loads that have been derived from dynamic analysis.

The pressure boundary portions of Class 1 valves in the RCS are designed and analyzed according to the requirements of NB-3500 of ASME III.

Valves in sample lines connected to the RCS are not considered to be ANS Safety Class 1 nor ASME Class 1. This is because the nozzles where the lines connect to the primary system piping are orificed to a 3/8 inch hole (see Section 5.4.3.2). This hole restricts the flow so that loss through a severance of one of these lines can be made up by normal charging flow.

3.9(N).1.5 Dynamic Analysis of Reactor Pressure Vessel for Postulated Loss-of-Coolant Accident

a. Introduction

This section presents the method of computing the reactor pressure vessel response to a postulated loss-of-coolant accident (LOCA). The structural analysis considers simultaneous application of the time-history loads on the reactor vessel resulting from the reactor coolant loop mechanical loads, and internal hydraulic pressure transients. The vessel is restrained by reactor vessel support pads and shoes beneath four of the reactor vessel nozzles and the reactor coolant loops with the primary supports of the steam generators and the reactor coolant pumps.

b. Interface Information

All input information was developed within Westinghouse. This information includes: reactor internals properties, loop mechanical loads and loop stiffnesses, internal hydraulic pressure transients, and reactor support stiffnesses. These inputs allowed formulation of the mathematical models performance of the analyses, as will be described.

c. Loading Conditions

Following a postulated pipe rupture at one of the three branch nozzle breaks per Subsection 3.6(N).2.3, the reactor vessel is excited by time-history forces. As previously mentioned, these forces are the combined effect of two phenomena: (1) reactor coolant loop mechanical loads, and (2) reactor internal hydraulic forces.

The reactor coolant loop mechanical forces are derived from the elastic analysis of the loop piping for the postulated break. This analysis is described in Subsection 3.9(N).1.4c. The loop mechanical forces which are released at the broken nozzle are applied to the vessel in the RPV blowdown analysis.

The reactor internals hydraulic pressure transients were calculated with the assumption that the structural motion is coupled with the pressure transients. This phenomenon has been referred to as hydroelastic coupling or fluid-structure interaction. The hydraulic analysis considers the fluid-structure interaction of the core barrel by accounting for the deflections of constraining boundaries which are represented by masses and springs. The dynamic response of the core barrel in its beam bending mode responding to blowdown forces compensates for internal pressure variation by increasing the volume of the more highly pressurized regions. The analytical methods used to develop the reactor internals hydraulics are described in WCAP-8708 (Reference 9).

d. Reactor Vessel and Internals Modeling

The reactor vessel is restrained by two mechanisms: (1) the four attached reactor coolant loops with the steam generator and reactor coolant pump primary supports and (2) four reactor vessel supports, two beneath reactor vessel inlet nozzles and two beneath reactor vessel outlet nozzles. The reactor vessel supports are described in Subsection 5.4.14 and are shown in Figure 5.4-14, and Figure 3.8-26. The support shoe provides restraint in the horizontal directions and for downward reactor vessel motion.

The reactor vessel model consists of two nonlinear elastic models connected at a common node. One model represents the dynamic vertical characteristics of the vessel and its internals, and the other model represents the translational and rotational characteristics of the structure. These two models are combined in the DARIWOSTAS code (Reference 1) to represent motion of the reactor vessel and its internals in the plane of the vessel centerline and the broken pipe centerline.

The model for horizontal motion considers that each node has one translational and one rotational degree of freedom in the vertical plane containing the centerline of the nozzle attached to the broken pipe and the centerline of the vessel. A combination of beam elements and concentrated masses are used to represent the components including the vessel, core barrel, neutron panels, fuel assemblies, and upper support columns. Connections between the various components are either pin-pin rigid links, translational impact springs with damping, or rotational springs.

The model for vertical motion considers that each mass node has one translational degree of freedom. The structure is represented by concentrated masses, springs, dampers, gaps, and frictional elements. The model includes the core barrel, lower support columns, bottom nozzles, fuel rods, top nozzles, upper support structure, and reactor vessel.

The horizontal and vertical models are coupled at the elevation of the primary nozzle centerlines. Node 1 of the horizontal model is coupled with node 2 of the vertical model at the reactor vessel nozzle elevation. This coupled node has external restraints characterized by a 3x3 matrix which represents the reactor coolant loop stiffness characteristics, by linear horizontal springs which describe the tangential resistance of the supports, and by individual nonlinear vertical stiffness elements which provide downward restraint only. The individual supports are located at the actual support pad locations and accurately represent the independent nonlinear behavior of each support.

e. Analytical Methods

The time-history effects of the internals loads and loops mechanical loads are combined and applied simultaneously to the appropriate nodes of the mathematical model of the reactor vessel and internals. The analysis is performed by numerically integrating the differential equations of motion to obtain the transient response. The output of the analysis includes the displacements of the reactor vessel and the loads in the reactor vessel supports which are combined with other applicable faulted condition loads and subsequently used to calculate the stresses in the supports. Also, the reactor vessel displacements are applied as a time-history input to the dynamic reactor coolant loop blowdown analysis. The resulting loads and stresses in the piping components and supports include both loop blowdown loads and reactor vessel displacements. Thus, the effect of vessel displacements upon loop response and the effect of loop blowdown upon vessel displacements are both evaluated.

f. Results of Analysis

As described, the reactor vessel and internals were analyzed for three postulated break locations. Table 3.9(N)-12 summarizes the allowable and no-loss-of-function displacements for reactor internals. Positive vertical displacement is up and positive horizontal displacement is away from and along the centerline of the vessel nozzle in the loop in which the break was postulated to occur. These displacements were calculated using an assumed break opening area for the postulated pipe ruptures of 144 in^2. Based on the deterministic fracture mechanics evaluation of the RCS loop piping, Westinghouse has demonstrated that postulation of pipe ruptures in the RC loop need not be made. An exemption from a portion of the requirements of General Design Criterion 4 of Appendix A to 10 CFR Part 50 has been granted to Seabrook; see Reference 14. The result of postulating rupture of one of the three branch line nozzles would be to impose reduced asymmetric loadings on the Reactor Core System. The fuel assembly grid load due to pipe ruptures was not applied to the analysis results.

The loads from the two sources, the internals reactions and the loop mechanical forces, are applied simultaneously in a nonlinear elastic dynamic time-history analysis on the model of the vessel, supports and internals. The results of this analysis are the dynamic loads on the reactor vessel supports and vessel time-history displacements. The maximum loads are combined with other applicable loads, such as seismic and deadweight and applied statically to the vessel support structure. The maximum stresses in the support are calculated and compared to faulted condition stress allowables given in Subsection 3.9(N).1.6.

3.9(N).1.6 Stress Criteria for Class 1 Components and Component Supports

All Class 1 components and supports are designed and analyzed for the Design, Normal, Upset, and Emergency Conditions to the rules and requirements of the ASME Code Section III. The design analysis or test methods and associated stress or load allowable limits that will be used in evaluation of Faulted Conditions are those that are defined in Appendix F of the ASME Code, as amended by R.G. 1.124 and variations stated in Updated FSAR Section 1.8, with supplementary options outlined below:

a. Elastic System Analysis and Component Inelastic Analysis

This is an acceptable method of evaluation for Faulted Conditions if the rules of F1323.1(a) are met for component supports, within the scope of Subsection NF and if primary stress limits for components are taken as greater of 0.70 S_u or S_y+1/3(S_u-S_y) for membrane stress and greater of 0.70 S_{ut} or S_y+ 1/3(S_{ut}-S_y) for membrane-plus-bending stress, where material properties are taken at appropriate temperature.

If plastic component analysis is used with elastic system analysis or with plastic system analysis, the deformations and displacements of the individual system members will be shown to be no larger than those which can be properly calculated by the analytical methods used for the system analysis.

b. Elastic/Inelastic System Analysis and Component/Test Load Method

The test load method given in F-1370(d) is an acceptable method of qualifying components in lieu of satisfying the stress/load limits established for the component analysis.

If the component/test load method is used with elastic or plastic system analysis, the deformations and displacements of the individual component members taken from the test load method data at the loads resulting from the system analysis will be shown to be no larger than those which can be properly calculated by the analytical methods used for the system analysis.

Loading combinations and allowable stresses for ASME III Class 1 components and supports are given in Table 3.9(N)-2 and Table 3.9(N)-3. For Faulted condition evaluations, the effects of the safe shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) are combined using the square-root-of-the-sum-of-the-squares (SRSS) method. Justification for this method of load combinations is contained in References 4 and 5.

3.9(N).1.7 Analytical Methods for RCS Class 1 Branch Lines

For faulted conditions analysis of Class 1 branch piping attached to the reactor coolant loop, Equation (9) of ASME III Subsection NB-3652 is applied with a stress limit of 3.0 S_m. This criterion provides sufficient assurance that the piping will not collapse or experience gross distortion such that the function of the system would be impaired. The basis for this position is described in Westinghouse response to NRC Question 110.34 on the RESAR-414 application (Docket No. STN 50-572), which subsequently received a preliminary design approval (PDA) in November 1978.

The analytical methods used to obtain the solution consist of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the response spectrum method for seismic dynamic analysis, and dynamic structural analysis for the effect of the three RCL branch nozzle breaks per Subsection 3.6(N).2.3.

The integrated Class 1 piping/supports system model is the basic system model used to compute loadings on components, component and piping supports, and piping. The system models include the stiffness and mass characteristics of the Class 1 piping components, the reactor coolant loop, and the stiffness of supports which affect the system response. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

a. Static

The Class 1 piping system models are constructed for the WESTDYN computer program, which numerically describes the physical system. A network model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the characteristics stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained.

After all the sections have been defined in this manner, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system. The support loads are also computed by multiplying the stiffness matrix by the displacement vector at the support point.

b. Seismic

The models used in the static analyses are modified for use in the dynamic analyses by including the mass characteristics of the piping and equipment.

The lumping of the distributed mass of the piping systems is accomplished by locating the total mass at points in the system which will appropriately represent the response of the distributed system. Effects of the primary equipment motion, that is, reactor vessel, steam generator, reactor coolant pump, and pressurizer, on the Class 1 piping system are obtained by modeling the mass and the stiffness characteristics of the primary equipment and loop piping in the overall system model.

The supports are represented by stiffness matrices in the system model for the dynamic analysis. Shock suppressors which resist rapid motions are also included in the analysis. The solution for the seismic disturbance employs the response spectra method. This method employs the lumped mass technique, linear elastic properties, and the principle of modal superposition.

The total response obtained from the seismic analysis consists of two parts: the inertia response of the piping system and the response from differential anchor motions. The stresses resulting from the anchor motions are considered to be secondary and, therefore, are included in the fatigue evaluation.

c. Loss-of-Coolant Accident

The mathematic models used in the seismic analyses of the Class 1 lines are also used for three RCL branch nozzle break effect analyses. To obtain the response for lines attached to the unaffected loops and lines attached to the affected loop, the deflections from the analysis of the reactor coolant loop are applied at branch nozzle connections.

d. Fatigue

A thermal transient heat transfer analysis is performed for each different piping component on all the Class 1 branch lines. The normal, upset, and test condition transients identified in Subsection 3.9.1.1 are considered in the fatigue evaluation.

The thermal quantities T_1, T_2, and a_aT_a, $-a_bT_b$ are calculated on a time-history basis, using a one-dimensional finite difference heat transfer computer program. Stresses due to these quantities were calculated for each time increment using the methods of NB-3650 of ASME III.

For each thermal transient, two loadsets are defined, representing the maximum and minimum stress states for that transient.

As a result of the normal mode spectral technique employed in the seismic analysis, the load components cannot be given signed values. Eight load sets are used to represent all possible sign permutations of the seismic moments at each point, thus insuring the most conservative combinations of seismic loads are used in the stress evaluation.

The WESTDYN computer program is used to calculate the primary-plus-secondary and peak stress intensity ranges, fatigue reduction factors and cumulative usage factors for all possible load set combinations. It is conservatively assumed that the transients can occur in any sequence, thus resulting in the most conservative and restrictive combinations of transients.

The combination of load sets yielding the highest alternating stress intensity range is determined and the incremental usage factor calculated. Likewise, the next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

3.9(N).2 Dynamic Testing and Analysis

3.9(N).2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

A preoperational piping vibrational and dynamics effects testing program will be conducted for the Reactor Coolant Loop/Supports System during pre-operational testing. The purpose of these tests will be to confirm that the system has been adequately designed and supported for vibration, as required by Section III of the ASME Code, paragraph NB-3622.3. The tests will include reactor coolant pump starts and trips. If vibration is experienced, which, from visual observation, appears to be excessive, either: (1) an instrumented test program on the piping, will be conducted and the system reanalyzed to demonstrate that the observed levels will not cause ASME Code stress and fatigue limits to be exceeded, (2) the cause of the excessive vibration will be eliminated, or (3) the support system will be modified to reduce the vibration. Particular attention will be provided at those locations where the vibration is expected to be the most severe for the particular transient condition being studied.

It should be noted that the layout, size, etc., of the reactor coolant loop and surge line piping used in the Seabrook plant is very similar to that employed in Westinghouse plants now in operation. The operating experience that has been obtained from these plants indicates that the reactor coolant loop and surge line piping are adequately designed and supported to minimize vibration. In addition, vibration levels of the reactor coolant pump, which is the only mechanical component that could cause vibration of the reactor coolant loop and surge line piping, are monitored as described in Subsection 5.4.1. Thus, excessive vibration of the reactor coolant loop and surge line piping should not be present. However, as added assurance that excessive piping vibration will not be present, the piping adjacent to the reactor coolant pump will be subjected to visual observation, as discussed above.

3.9(N).2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

The operability of seismic Category I mechanical equipment must be demonstrated if the equipment is active, i.e., ASME Section III components that must perform a mechanical motion during the course of performing their safety function in shutting down the plant to and maintaining it in a cold shutdown or in mitigating the consequences of a postulated event. The operability of active Safety Class 2 and 3 pumps, active Safety Class 1, 2 or 3 valves and their respective drives, operators and vital auxiliary equipment is shown by satisfying the criteria given in Subsection 3.9(N).3.2. Other active mechanical equipment is shown operable by either testing, analysis or a combination of testing and analysis. The operability programs implemented on this other active equipment are similar to the program described in Subsection 3.9(N).3.2 for pumps and valves. Testing procedures similar to the procedures outlined in Section 3.10(N) for electrical equipment are used to demonstrate operability if the component is mechanically or structurally complex such that its response cannot be adequately predicted analytically. Analysis may be used if the equipment is amenable to modeling and dynamic analysis.

Inactive seismic Category I equipment such as heat exchangers, racks and consoles are shown to have structural integrity during a seismic event by one of the following methods: (1) by analysis satisfying the stress criteria applicable to the particular piece of equipment, or (2) by test showing that the equipment retains its structural integrity under the simulated test environment.

As indicated above, Westinghouse used analysis, test, or a combination of test and analysis to seismically qualify originally-designated active equipment. Testing was the preferred method. However, analysis was used when one of the following conditions was satisfied:

a. The equipment is too large or the external loads, connecting elements, or appurtenances cannot be simulated with a shaker table test.

b. The only requirement that must be satisfied relative to the safety of the plant is the maintenance of structural integrity (mechanical equipment only).

c. The component represents a simple linear system, or nonlinearities can be conservatively accounted for in the analysis.

The seismic qualification of electrical equipment by analysis was not employed by Westinghouse. However, analysis was employed to supplement tests or to provide verification that test results are applicable for a particular configuration. Components not originally-designated as active were reviewed for loading and seismic accelerations to ensure that they meet active component requirements.

A list of seismic Category I equipment is provided in Table 3.2-1 and Table 3.2-2.

3.9(N).2.3 Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions

The vibration characteristics and behavior due to flow-induced excitation are very complex and not readily ascertained by analytical means alone. Reactor components are excited by the flowing coolant which causes oscillatory pressures on the surfaces. The integration of these pressures over the applied area should provide the forcing functions to be used in the dynamic analysis of the structures. In view of the complexity of the geometries and the random character of the pressure oscillations, a closed form solution of the vibratory problem by integration of the differential equation of motion is not always practical and realistic. The determination of the forcing functions as a direct correlation of pressure oscillations cannot be practically performed independently of the dynamic characteristics of the structure. The main objective is to establish the characteristics of the forcing functions that essentially determine the response of the structures. By studying the dynamic properties of the structure from previous analytical and experimental work, the characteristics of the forcing function can be deduced. These studies indicated that the most important forcing functions are flow turbulence, and pump-related excitation. The relevance of such excitation depends on many factors, such as type and location of component and flow conditions. The effects of these forcing functions have been studied from tests performed on models and prototype plants, as well as component tests (References 6, 7 and 8).

The Indian Point No. 2 plant has been established as the prototype for a four-loop plant internals verification program, and was fully instrumented and tested during hot functional testing. In addition, the Trojan plant instrumentation program provides, and the Sequoyah No. 1 instrumentation program will provide, prototype data applicable to Seabrook (References 7 and 8).

Seabrook Station is similar to Indian Point No. 2; the only significant differences are the modifications resulting from the use of 17x17 fuel, replacement of the annular thermal shield with neutron shielding panels, and the change to the UHI-style inverted top hat support structure configuration. These differences are addressed below.

a. 17x17 Fuel

The only structural changes in the internals resulting from the design change from the 15x15 to the 17x17 fuel assembly are the guide tube and control rod drive line. The new 17x17 guide tubes are stronger and more rigid, hence they are less susceptible to flow-induced vibration. The fuel assembly itself is relatively unchanged in mass and spring rate, and thus no significant deviation is expected from the 15x15 fuel assembly vibration characteristics.

b. Neutron Shielding Pads Lower Internals

The primary cause of core barrel excitation is flow turbulence generated in the downcomer annulus (Reference 7). The vibration levels due to core barrel excitation for Trojan and Seabrook, both having neutron shielding pads, are expected to be similar. Since Seabrook has greater velocities than Trojan, vibration levels due to the core barrel excitation is expected to be somewhat greater than that for Trojan (proportional to flow velocity raised to a smaller power) (Reference 6). However, scale model test results (Reference 6) and results from Trojan (Reference 8) show that core barrel vibration of plants with neutron shielding pads is significantly less than that of plants with thermal shields. This information, and the fact that low core barrel flange stresses with large safety margins were measured at Indian Point No. 2 (thermal shield configuration), lead to the conclusion that stresses approximately equal to those of Indian Point No. 2 will result on the Seabrook internals, with the attendant large safety margins.

c. UHI-Style Inverted Top Hat Upper Support Configuration

The components of the upper internals are excited by turbulent forces due to axial and cross flows in the upper plenum (Reference 7). Sequoyah and Seabrook have the same basic upper internals configuration, therefore, the general vibration behavior is not changed. Since Seabrook has slightly higher velocities than Sequoyah, the Seabrook upper internals vibration is expected to be slightly greater than that for Sequoyah. However, scale-model test results and analytical work predict high safety margins for the Sequoyah upper internals components. Therefore, the Sequoyah results can be used quite confidently to demonstrate the adequacy of the Seabrook upper internals.

The original test and analysis of the four-loop configuration is augmented (References 6, 7 and 8) to cover the effects of successive hardware modifications.

3.9(N).2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Because the Seabrook reactor internals design configuration is well characterized, as was discussed in Subsection 3.9(N).2.3, it is not considered necessary to conduct instrumented tests of the Seabrook plant hardware. The recommendations of Regulatory Guide 1.20 will be met by conducting the confirmatory preoperational testing examination for integrity. This examination will include approximately 30 features, with special emphasis on the following areas:

a. All major load-bearing elements of the reactor internals relied upon to retain the core structure in place

b. The lateral, vertical and torsional restraints provided within the vessel

c. Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals

d. Those other locations on the reactor internal components which are similar to those which were examined on the prototype designs

e. The inside of the vessel will be inspected before and after the hot functional test, with all the internals removed, to verify that no loose parts of foreign material are in evidence.

A particularly close inspection will be made on the following items or areas using a 5X or 10X magnifying glass, or other appropriate inspection.

a. Lower Internals

1. Upper barrel to flange girth weld.

2. Upper barrel to lower barrel girth weld.

3. Upper core plate aligning pin. Examine bearing surfaces for shadow marks, burnishing, buffing or scoring. Inspect welds for integrity.

4. Irradiation specimen guide screw locking devices and dowel pins. Check for lockweld integrity.

5. Baffle assembly locking devices. Check for lockweld integrity.

6. Lower barrel to core support girth weld.

7. Neutron shielding pad screw locking devices and dowel pin lock welds. Examine the interface surfaces for evidence of tightness. Check for lockweld integrity.

8. Radial support key welds.

9. Insert screw locking devices. Examine soundness of lockwelds.

10. Core support columns and instrumentation guide tubes. Check the joints for tightness and soundness of the locking devices.

11. Secondary core support assembly screw locking devices for lockweld integrity.

12. Lower radial support keys and inserts. Examine bearing surfaces for shadow marks, burnishing, buffing or scoring. Check the integrity of the lockwelds. These members supply the radial and torsional constraint of the internals at the bottom relative to the reactor vessel while permitting axial and radial growth between the two. Subsequent to the hot functional testing, the bearing surfaces of the key and keyway will show burnishing, buffing or shadow marks which indicate pressure loading and relative motion between these parts. Minor scoring of engaging surfaces is also possible and acceptable.

13. Gaps at baffle joints. Check gaps between baffle to baffle joints.

b. Upper Internals

1. Guide tube, support column and orifice plate locking devices.

2. Support column welds.

3. Upper core plate alignment inserts. Examine for shadow marks, burnishing, buffing or scoring. Check the locking devices for integrity of lockwelds.

4. Guide tube enclosure welds, tube-transition plate welds and card welds.

Acceptance standards are the same as required in the shop by the original design drawings and specifications.

During the hot functional test, the internals will be subjected to a total operating time at greater than normal full-flow conditions (four pumps operating) of at least 240 hours. This provides a cyclic loading of approximately 10^7 cycles on the main structural elements of the internals. In addition, there will be some operating time with only one, two and three pumps operating.

Pre- and post-hot-functional inspection results serve to confirm that the internals are well behaved. When no signs of abnormal wear or harmful vibrations are detected, and no apparent structural changes take place, the four-loop core support structures are considered to be structurally adequate and sound for operation.

3.9(N).2.5 Dynamic System Analysis of the Reactor Internals under Faulted Conditions

Analysis of the reactor internals for blowdown loads resulting from a loss-of-coolant accident is based on the time-history response of the internals to simultaneously applied blowdown forcing functions. The forcing functions are defined at points in the system where changes in cross section or direction of flow occur, such that differential loads are generated during the blowdown transient. The dynamic mechanical analysis can employ the displacement method, lumped parameters, stiffness matrix formulations, and assumes that all components behave in a linearly elastic manner.

In addition, because of the complexity of the system and the components, it is necessary to use finite element stress analysis codes to provide more detailed information at various points.

A computer program, MULTIFLEX (Reference 9 and 10), which was developed to calculate local fluid pressure, flow, and density transients that occur in pressurized water reactor coolant systems during a loss-of-coolant accident is applied to the subcooled, transition, and saturated two-phase blowdown regimes. This MULTIFLEX code is based on the method of characteristics wherein the resulting set of ordinary differential equations, obtained from the laws of conservation of mass, momentum, and energy are solved numerically using a fixed mesh in both space and time.

Although spatially one-dimensional conservation laws are employed, the code can be applied to describe three-dimensional system geometries by use of the equivalent piping networks. Such piping networks may contain any number of pipes or channels of various diameters, dead ends, branches (with up to six pipes connected to each branch), contractions, expansions, orifices, pumps and free surfaces (such as in the pressurizer). System losses such as friction, contraction, expansion, etc., are considered.

The reanalysis performed in support of increased core power has made use of the MULTIFLEX (Reference 9 and 10) computer code. MULTIFLEX is an extension of the BLODWN-2 computer code and includes mechanical structure models and their interactions with the thermal-hydraulic system. Both versions of the MULTIFLEX code share a common hydraulic modeling scheme, with differences being confined to a more realistic downcomer hydraulic network and a more realistic core barrel structural model that accounts for non-linear boundary conditions and vessel motion. Generally, this improved modeling results in lower, more realistic, but still conservative hydraulic forces on the core barrel. The NRC staff has accepted (Reference 15) the use of MULTIFLEX 3.0 for calculating the hydraulic forces on reactor vessel internals (Reference 16).

The MULTIFLEX code evaluates the pressure and velocity transients throughout the system. These pressure and velocity transients are stored as a file and are made available to the programs LATFORC and FORCE2, which use detailed geometric descriptions to calculate the loadings on the reactor internals.

The LATFORC code (Appendix A of Reference 9) computes the horizontal or lateral forces on the reactor vessel shell interior, the core barrel, and the neutron pads due to the pressure on the location at up to ten elevations.

The FORCE2 code (Appendix B of Reference 9) computes the vertical forces on the reactor internals components. Each reactor component for which FORCE2 calculations are required is designated as an element and assigned an element number. Forces acting on each of the elements are calculated by summing the effects of:

a. The pressure differential across the element.

b. Flow stagnation on, and unrecovered orifice losses across the element.

c. Friction losses along the element.

Input to the code, in addition to the blowdown pressure and velocity transients, includes the effective area of each element on which the force acts due to the pressure differential across the element, a coefficient to account for flow stagnation and unrecovered orifice losses, and the total area of the element along which shear forces act.

The mechanical analysis (Reference 11) has been performed using conservative assumptions. Some of the more significant assumptions are:

a. The mechanical and hydraulic analyses have considered the effect of hydroelasticity.

b. The reactor internals are represented by a multi-mass system connected with springs and dashpots simulating the elastic response and the viscous damping of the components. The modeling is conducted in such a way that uniform masses are lumped into easily identifiable discrete masses while elastic elements are represented by springs.

The model described is considered to have a sufficient number of degrees of freedom to represent the most important modes of vibration in the vertical direction. This model is conservative in the sense that further mass-spring resolution of the system would lead to further attenuation of the shock effects obtained with the present model.

The pressure waves generated within the reactor are highly dependent on the location and nature of the postulated pipe failure. In general, the more rapid the severance of the pipe, the more severe the imposed loadings on the components. A one millisecond severance time is taken as the limiting case.

In the case of the hot leg branch line break, the vertical hydraulic forces produce an initial upward lift of the core. A rarefaction wave propagates through the reactor hot leg nozzle into the interior of the upper core barrel. Since the wave has not reached the flow annulus on the outside of the barrel, the upper barrel is subjected to an impulsive compressive wave. Thus, dynamic instability (buckling) or large deflections of the upper core barrel, or both, is a possible response of the barrel during a hot leg branch line break, and results in transverse loading on the upper core components as the fluid exits the hot leg nozzle.

In the case of the cold leg branch line break, a rarefaction wave propagates along a reactor inlet pipe, arriving first at the core barrel at the inlet nozzle of the broken loop. The upper barrel is then subjected to a non-axisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel. After the cold leg branch line break, the initial steady-state hydraulic lift forces (upward) decrease rapidly (within a few milliseconds) and then increase in the downward direction. These cause the reactor core and lower support structure to move initially downward.

If a simultaneous seismic event with the intensity of the SSE is postulated with the loss-of-coolant accident, the imposed loading on the internals component is additive and, therefore, the combined loading is considered even though the loading imposed by the earthquake is generally small compared to the blowdown loading.

The summary of the mechanical analysis follows:

a. Mathematical Model of the Reactor Pressure Vessel and Analytical Method

The mathematical model of the RPV is a three-dimensional nonlinear finite element model which represents the dynamic characteristics of the reactor vessel and its internals in the six geometric degrees of freedom. The model was developed using the WECAN computer code. The model consists of three concentric structural submodels connected by nonlinear impact elements and stiffness matrices. The first submodel represents the reactor vessel shell and associated components. The reactor vessel is restrained by four reactor vessel supports (situated beneath alternate nozzles) and by the attached primary coolant piping. Each reactor vessel support is modeled by a linear horizontal stiffness and a vertical spring-gap element. The attached piping is represented by a stiffness matrix.

The second submodel represents the reactor core barrel (RCB), neutron panels, lower support plate, tie plates, and secondary core support components. This submodel is physically located inside the first, and is connected to it by a stiffness matrix at the internals support ledge. Core barrel to vessel shell impact is represented by nonlinear elements at the core barrel flange, core barrel nozzle, and lower radial support locations.

The third and innermost submodel represents the upper support plate, guide tubes, upper support columns, upper and lower core plates, and fuel. The third submodel is connected to the first and second by stiffness matrices and nonlinear elements.

Fluid-structure or hydro-elastic interaction is included in the reactor pressure vessel model for seismic evaluation. The horizontal hydro-elastic interaction is significant in the cylindrical fluid flow region between the core barrel and reactor vessel (the downcomer). Mass matrices with off-diagonal terms (horizontal degrees-of-freedom only) attach between nodes on the core barrel and reactor vessel shell.

The diagonal terms of the mass matrix are similar to the lumping of water mass to the vessel shell and core barrel. The off-diagonal terms reflect the fact that all the water mass does not participate when there is no relative motion of the vessel and core barrel. It should be pointed out that the hydrodynamic mass matrix has no artificial virtual mass effect and is derived in a straight-forward, quantitative manner.

The matrices are a function of the properties of two cylinders with a fluid in the cylindrical annulus, specifically; inside and outside radius of the annulus, density of the fluid and length of the cylinders. Vertical segmentation of the RCB allows inclusion of radii variations along the RCB height and approximates the effects of RCB beam deformation. These mass matrices were inserted between selected nodes on the core barrel and reactor vessel shell.

For LOCA the time-history effects of the internals loads and loop mechanical loads are combined and applied simultaneously to the appropriate nodes of the mathematical model of the reactor vessel and internals. The analysis is performed by numerically integrating the differential equations of motion to obtain the transient response. The output of the analysis includes the displacements of the reactor vessel and the reactor internals.

b. Transverse Excitation Model for Blowdown

Various reactor internal components are subjected to transverse excitation during blowdown. Specifically, the barrel, guide tubes, and upper support columns are analyzed to determine their response to this excitation.

c. Core Barrel

For the hydraulic analysis of the pressure transients during hot leg blowdown, the maximum pressure drop across the barrel is a uniform radial compressive impulse.

The barrel is then analyzed for dynamic buckling using the following conservative assumptions:

1. The effect of the fluid environment is neglected

2. The shell is treated as simply supported.

During cold leg blowdown, the upper barrel is subjected to a non-axisymmetric expansion radial impulse which changes as the rarefaction wave propagates both around the barrel and down the outer flow annulus between vessel and barrel.

The analysis of transverse barrel response to cold leg blowdown is performed as follows:

1. The core barrel is treated as a simply supported cylindrical shell of constant thickness between the upper flange weldment and the lower support plate weldment. No credit is taken for the supports at the barrel midspan offered by the outlet nozzles. This assumption leads to conservative deflection estimates of the upper core barrel.

2. The core barrel is analyzed as a shell with two variable sections to model the support flange and core barrel.

3. The barrel with the core and neutron shielding pads is analyzed as a beam fixed at the top and elastically supported at the lower radial support and the dynamic response is obtained.

d. Guide Tubes

The guide tubes in closest proximity to the outlet nozzle of the affected loop are the most severely loaded during a blowdown. The transverse guide tube forces decrease with increasing distance from the ruptured nozzle location.

All of the guide tubes are designed to maintain the function of the control rods for a break size of 144 in^2 and smaller. No credit for the function of the control rods is assumed for break size areas above 144 in^2. However, the design of the guide tube will permit control rod operation in all but four control rod positions, which is sufficient to maintain the core in a subcritical configuration, for break sizes up to a double-ended hot leg break. This double-ended hot leg break imposes the limiting lateral guide tube loading.

e. Upper Support Columns

Upper support columns located close to the broken nozzle during hot leg branch line break will be subjected to transverse loads due to cross flow. The loads applied to the columns are computed with a method similar to the one used for the guide tubes, i.e., by taking into consideration the increase in flow across the column during the accident. The columns are studied as beams with variable section, and the resulting stresses are obtained using the reduced section modulus and appropriate stress risers for the various sections.

The effects of the gaps that could exist between vessel and barrel, between fuel assemblies, and between fuel assemblies and baffle plates are considered in the analysis. Linear analysis will not provide information about the impact forces generated when components impinge each other, but can, and is, applied prior to gap closure. Reference 11 provides further details of the blowdown method used in the analysis of the reactor internals.

The stresses due to the safe shutdown earthquake (vertical and horizontal components) are combined with the blowdown stresses to obtain the largest principal stress and deflection.

f. Results

All reactor internals components were found to be within acceptable stress and deflection limits for both hot leg and cold leg loss-of-coolant accidents, occurring simultaneously with the safe shutdown earthquake.

The results obtained from the linear analysis indicate that during blowdown, the relative displacement between the components will close the gaps and, consequently, the structures will impinge on each other, making the linear analysis unrealistic and forcing the application of nonlinear methods to study the problem. Although linear analysis will not provide information about the impact forces generated when components impinge on each other, it can, and is, applied prior to gap closure. The effects of the gaps that could exist between vessel and barrel, between fuel assemblies, between fuel assemblies and baffle plates, and between the control rods and their guide paths were considered in the analysis. Both static and dynamic stress intensities are within acceptable limits. In addition, the cumulative fatigue usage factor is also within the allowable usage factor of unity.

The stresses due to the Safe Shutdown Earthquake (vertical and horizontal components) were combined in the most unfavorable manner with the blowdown stresses to obtain the largest principal stress and deflection.

These results indicate that the maximum deflections and stresses in the critical structures are below the established allowable limits. For the transverse excitation, it is shown that the upper barrel does not buckle during a hot leg break, and that it has an allowable stress distribution during a cold leg break.

Even though control rod insertion is not required for plant shutdown, this analysis shows that most of the guide tubes will deform within the limits established experimentally to assure control rod insertion. For the guide tubes deflected above the no-loss-of-function limit, it must be assumed that the rods will not drop. However, the core will still shut down due to the negative reactivity insertion in the form of core voiding. Shutdown will be aided by the great majority of rods that do drop. Seismic deflections of the guide tubes are generally negligible by comparison with the no-loss-of-function limit.

3.9(N).2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

As stated in Subsection 3.9(N).2.3, it is not considered necessary to conduct instrumented tests of the Seabrook reactor vessel internals. Adequacy of these internals will be verified by use of the Sequoyah and Trojan results. References 7 and 8 describe predicted vibration behavior based on studies performed prior to the plant tests. These studies, which use analytical models, scale model test results, component tests, and results of previous plant tests, are used to characterize the forcing functions and establish component structural characteristics, so that the flow induced vibratory behavior and response levels for Seabrook are estimated. These estimates are then compared to values deduced from plant test data obtained from the Sequoyah and the Trojan internals vibration measurement programs.

3.9(N).3 ASME Code Class 1, 2 and 3 Components, Component Supports and Core Support Structures

The ASME Code Class components are constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III.

Detailed discussion of ASME Code Class 1 components is provided in Section 5.4 and Subsection 3.9(N).1. For core support structures, design loading conditions are given in Subsection 3.9(N).5.

In general, for reactor internals components and for core support structures, the criteria for acceptability in regard to mechanical integrity analyses are that adequate core cooling and core shutdown must be assured. This implies that the deformation of the reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections and stability of the parts, in addition to a stress criterion to assure integrity of the components.

For the loss of coolant plus the safe shutdown earthquake condition, deflections of critical internal structures are limited. In a hypothesized downward vertical displacement of the internals, energy-absorbing devices limit the displacement after contacting the vessel bottom head, ensuring that the geometry of the core remains intact.

The following mechanical functional performance criteria apply:

a. Following the loss-of-coolant accident, the functional criterion to be met for the reactor internals is that the plant can be shutdown and cooled in an orderly fashion, so that fuel cladding temperature is kept within specified limits. This criterion implies that the deformation of critical components must be kept sufficiently small to allow core cooling.

b. For large reactor coolant branch nozzle pipe breaks, the reduction in water density greatly reduces the reactivity of the core, thereby shutting down the core whether the control rods are tripped or not. The subsequent refilling of the core by the Emergency Core Cooling System uses borated water to maintain the core in a subcritical state. Therefore, the main requirement is to assure effectiveness of the Emergency Core Cooling System. Insertion of the control rods, although not needed, gives further assurance of ability to shut the plant down and keep it in a safe shutdown condition.

c. The inward upper barrel deflections are controlled to insure no contacting of the nearest rod cluster control guide tube. The outward upper barrel deflections are controlled in order to maintain an adequate annulus for the coolant between the vessel inner diameter and core barrel outer diameter.

d. The rod cluster control guide tube deflections are limited to insure operability of the control rods.

e. To insure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited.

Methods of analysis and testing for core support structures are discussed in Subsections 3.9(N).2.3, 3.9(N).2.5 and 3.9(N).2.6. Stress limits and deformation criteria are given in Subsection 3.9(N).5.

3.9(N).3.1 Loading Combinations, Design Transients, and Stress Limits (for ASME Code Class 2 and 3 Components)

Design pressure, temperatures, and other loading conditions that provide the bases for design of fluid systems Code Class 2 and 3 components are presented in the sections which describe the systems.

a. Design Loading Combinations

The design loading combinations for ASME Code Class 2 and 3 components and supports are given in Table 3.9(N)-4. The design loading combinations are categorized with respect to Normal, Upset, Emergency, and Faulted Conditions. Stress limits for each of the loading combinations are component-oriented and are presented in Table 3.9(N)-5, Table 3.9(N)-6, Table 3.9(N)-7, Table 3.9(N)-8 and Table 3.9(N)-9 for tanks, inactive (operability is not relied upon to perform a safety function as defined for active in Section 3.9(N).2.2) pumps, active (operability is relied upon to perform a safety function as defined for active in Section 3.9(N).2.2) pumps, and valves, respectively. Active pumps and valves are discussed in Subsection 3.9(N).3.2. Design of component supports is discussed in Subsection 3.9(N).3.4.

b. Design Stress Limits

The design stress limits established for the components are sufficiently low to assure that violation of the pressure-retaining boundary will not occur. These limits, for each of the loading combinations, are component-oriented and are presented in Table 3.9(N)-5, Table 3.9(N)-6, Table 3.9(N)-7, Table 3.9(N)-8 and Table 3.9(N)-9.

3.9(N).3.2 Pump and Valve Operability Assurance

a. Pump and Valve Operability Program

Mechanical equipment classified as safety-related must be capable of performing its function under postulated plant conditions. Equipment with faulted condition functional requirements includes active pumps and valves in fluid systems important to safety. Seismic analysis is presented in Section 3.7 and covers all safety-related mechanical equipment. A list of all active pumps supplied by Westinghouse is presented in Table 3.9(N)-10. Active valves supplied by Westinghouse are listed in Table 3.9(N)-11. The Westinghouse requirements necessary to implement the operability program described below are contained in the appropriate equipment specifications for active pumps and valves.

1. <u>Pump Operability Program</u>

All active pumps 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 150 percent of the design pressure, (2) seal leakage tests at the same pressure used in the hydrostatic tests, and (3) performance tests, while the pump is operated with flow, to determine total developed head, minimum and maximum head, Net Positive Suction Head (NPSH) requirements and other pump/motor parameters. Also monitored during these operating tests are bearing temperatures and vibration levels. Bearing temperature limits are determined by the manufacturer, based on the bearing material, clearances, oil type, and rotational speed.

After the pump is installed in the plant, it undergoes cold hydro tests, hot functional tests, and the required periodic in-service inspection and operation. These tests demonstrate that the pump will function as required during all normal operating conditions for the design life of the plant.

In addition to these tests, the safety-related active pumps are qualified for operability during an SSE condition by assuring that the pump will continue operating and will not be damaged during the seismic event.

The pump manufacturer is required to show that the pump will operate normally when subjected to the maximum seismic accelerations and maximum faulted nozzle loads. It is required that tests or dynamic analysis be used to show that the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, is considered rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor is performed with the conservative SSE accelerations of 2.1g in two orthogonal horizontal directions and 2.1g vertical acceleration acting simultaneously. The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. The nature of seismic disturbances dictates that the maximum contact (if it occurs) will be of short duration. 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 as indicated in Table 3.9(N)-8. In addition, the pump casing stresses caused by the maximum seismic nozzle loads are limited to stresses outlined in Table 3.9(N)-8. The maximum seismic nozzle loads are 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(N)-8 as allowables assures that critical parts of the pump will not be damaged during the faulted condition and that, therefore, the reliability of the pump for post-faulted condition operation will not be impaired by the seismic event.

The second criterion necessary to assure operability is that the pump will function throughout the SSE. The pump/motor combination is designed to rotate at a constant speed under all conditions unless the rotor becomes completely seized, i.e., with no rotation. Typically, the rotor can be seized 5 full seconds before a circuit-breaker, to prevent damage to the motor, shuts down the pump. However, the high rotary inertia in the operating pump rotor, and the nature of the random, short duration loading characteristics of the seismic event, will prevent the rotor from losing its function. 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 SSE and will operate at the design speed despite the SSE loads.

To complete the seismic qualification procedures, the pump motor is independently qualified for operation during the maximum seismic event. Any auxiliary equipment which is identified to be vital to the operation of the pump or pump motor, and which is not qualified for operation during the pump analysis or motor qualifications, is also separately qualified for operation at the accelerations it would see at its mounting. The pump motor and vital auxiliary equipment is qualified by meeting the requirements of IEEE Standard 344-1975, with the additional requirements and justifications outlined in Subsection 3.9(N).3.2b.

The program above gives the required assurance that the safety-related pump/motor assemblies will not be damaged and will continue operating under SSE loadings, and, therefore, 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.

Since the pump is not damaged during the faulted condition, the functional ability of active pumps after the faulted condition is assured since only normal operating loads and steady-state nozzle loads exist.

Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads will be identical to the normal plant operating loads. 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.

2. Valve Operability Program

Active valves are subjected to a series of tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test to ASME Section III requirements, backseat and main seat leakage tests, disc hydrostatic test, and operational tests to verify that the valve will open and close. Qualification of motor operators for environmental conditions is discussed in Section 3.11(N). Cold hydro tests, hot functional qualification tests, periodic in-service inspections, and periodic in-service operations in accordance with the IST program, or other Station procedures or Technical Specification surveillances are performed in situ to verify and assure the functional ability of the valve. These tests guarantee reliability of the valve for the design life of the plant.

The valves are constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The maximum stress limits used for active Class 2 and 3 valves are shown in Table 3.9(N)-9. On active valves, an analysis of the extended structure is performed for static equivalent seismic loads applied at the center of gravity of the extended structure.

In addition to these tests and analyses, representative valves of each design type that were originally designated as active valves were tested for verification of operability during a simulated plant faulted condition event by demonstrating operational capabilities within the specified limits. The testing procedures are described below.

The valve was mounted in a manner which conservatively represents typical valve installations. The valve included the operator pilot solenoid valves, and limit switches when such are normally attached to the valve in service. The faulted condition nozzle loads were limited so that the operability of the valve was not affected. The operability of the valve during a faulted condition was demonstrated by satisfying the following criteria:

(a) Active valves are designed to have a first natural frequency which is greater than 33 Hz. This may be shown by test or analysis. For valves with a fundamental frequency below 33 Hz, the valve flexibility shall be accounted for in the subsystem analysis.

(b) The actuator and yoke of the valve system was statically deflected an amount equal to the deflection caused by the faulted condition accelerations applied at the center of gravity of the extended structure alone in the direction of the weakest axis of the yoke. The design pressure of the valve was applied to the valve during the static deflection tests.

(c) The valve was cycled while in the deflected position. The time required to open or close the valve in the deflected position was compared to similar design taken in the undeflected condition to evaluate the significance of any change.

(d) Motor operators, external limit switches, and pilot solenoid valves necessary for operation are qualified in accordance with IEEE Standard 344-1975, with the additional requirements and justifications as supplied in Subsection 3.9(N).3.2b.

The accelerations which are used for the static valve qualification shall be equivalent, as justified by analysis, to the simultaneous application of at least 2.1g in two orthogonal horizontal directions and at least 2.1g in the vertical direction. The piping is designed to maintain the operator accelerations to these levels.

The above testing program applied to valves with extended structures. The testing was conducted on a representative number of originally-designated active valves. Valves from each of the primary safety-related design types were tested. Valve sizes which cover the range of sizes in service were qualified by the tests and the results are used to qualify all valves within the intermediate range of sizes.

Valves which are safety-related but can be classified as not having an extended structure, such as check valves and safety valves, are considered separately.

Check valves are characteristically simple in design and their operation will not be affected by seismic accelerations or the maximum applied nozzle loads. The check valve design is compact and there are no extended structures or masses whose motion could cause distortions which could restrict operation of the valve. The nozzle loads due to maximum seismic excitation will not affect the functional ability of the valve since the valve disc is typically designed to be isolated from the body wall. The clearance supplied by the design around the disc will prevent the disc from becoming bound or restricted due to any body distortions caused by nozzle loads. Therefore, the design of these valves is such that once the structural integrity of the valve is assured using standard methods, the ability of the valve to operate is assured by the design features. The valve will also undergo the following: (1) in shop hydrostatic test, (2) in shop seat leakage test, and (3) periodic in situ valve exercising and inspection to assure the functional ability of the valve.

The pressurizer safety valves are qualified by the following procedures: (1) stress and deformation analyses of critical items which may affect operability for faulted condition loads, (2) in shop hydrostatic and seat leakage tests, and (3) periodic in situ valve inspection. In addition to these tests, a static load equivalent to that applied by the faulted condition is applied at the top of the bonnet and the pressure is increased until the valve mechanism actuates. Successful actuation within the design requirements of the valve assures its overpressurization safety capabilities during a seismic event.

Using these methods, active valves are qualified for operability during a faulted event. These methods outlined above conservatively simulate the seismic event and assure that the active valves will perform their safety-related function when necessary.

b. <u>Pump Motor and Valve Operator Qualification</u>

Active pump motors (including vital pump appurtenances) and active valve motor operators are seismically qualified in accordance with IEEE Standard 344-1975. The seismic qualification program for this electrical equipment is further described in Section 3.10(N) and the Equipment Qualification Data Packages referenced therein.

3.9(N).3.3 <u>Mounting of Pressure Relief Devices</u>

Refer to Subsection 3.9(B).3.3.

3.9(N).3.4 Component Supports (ASME Code Class 2 and 3)

See Subsection 3.9(N).1 for ASME Code Class 1 component supports.

a. Component Supports for Components Procured After July 1, 1974

Class 2 and 3 component supports are designed and analyzed for Design, Normal, Upset, Emergency, and Faulted conditions to the rules and requirements of Subsection NF of Section III of the ASME B&PV Code (1974 Edition). The design analyses or test methods and associated stress or load allowable limits that are used in the evaluation of linear supports for Faulted conditions are those defined in Appendix F of the ASME Code.

b. Component Supports for Components Procured Prior to July 1, 1974

Class 2 and 3 supports are designed as follows:

1. Standard Component Supports

(a) Normal - The allowable stresses or load ratings of MSS-SP-58 are used.

(b) Upset - For upset conditions, the allowable stresses or load ratings are 20 percent higher than those specified for normal conditions.

(c) Emergency - For emergency conditions, the allowable stresses or load ratings are 80 percent higher than those specified for normal conditions. Supports (rod hangers and struts) are checked for elastic stability when applicable, and maximum compressive load does not exceed critical buckling load as specified by the applicable codes and design standards.

(d) Faulted - The allowable stresses or load ratings of MSS-SP-58 are based on a factor of safety which is greater than or equal to four, i.e., the allowable stress is less than or equal to one-fourth the minimum tensile stress of the material. The allowable stresses for faulted conditions are thus less than or equal to 0.6 times the minimum tensile stress of the material, i.e., 2.4 times one-fourth the minimum tensile stress of the material equals 0.6 times the minimum tensile stress. This low allowable stress (associated factor of safety equals 1.67) is adequate to assure that active components are properly supported for faulted conditions.

2. Linear Type Supports

(a) Normal - The allowable stresses of AISC-69 Part 1 are employed for normal condition limits.

(b) Upset - Stress limits for upset conditions are 33 percent higher than those specified for normal conditions. This is consistent with paragraph 1.5.6 of AISC-69 Part 1 which permits one-third increase in allowable stresses for wind or seismic loads.

(c) Emergency - Not applicable.

(d) Faulted - Stress limits for faulted condition are the same as for the upset condition.

3. Plate and Shell Type Supports

(a) Normal - Normal condition limits are those specified in ASME Section VIII Division 1 or AISC-69 Part 1.

(b) Upset - Stress limits for upset conditions are 33 percent higher than those specified for normal conditions. This is consistent with paragraph 1.5.6 of AISC Part 1 which permits one-third increase in allowable stresses for wind or seismic loads.

(c) Emergency - Not applicable.

(d) Faulted - Stress limits for faulted condition are the same as for the upset condition.

3.9(N).4 Control Rod Drive System (CRDS)

3.9(N).4.1 Descriptive Information of CRDS

a. Control Rod Drive Mechanisms

Control Rod Drive Mechanisms (CRDMs) are located on the head of the reactor vessel. They are coupled to Rod Cluster Control Assemblies which contain neutron absorber material over the entire length of the control rods. The control rod drive mechanism is shown in Figure 3.9(N)-3 and schematically in Figure 3.9(N)-4.

The primary function of the CRDM is to insert or withdraw the Rod Cluster Control Assemblies within the core to control average core temperature and to shut down the reactor.

The CRDM is a magnetically operated jack, which is an arrangement of three electromagnets energized in a controlled sequence by a power cycler, to insert or withdraw Rod Cluster Control Assemblies in the reactor core in discrete steps. Rapid insertion of the Rod Cluster Control Assemblies occurs when electrical power is interrupted.

The CRDM consists of four separate subassemblies. They are the pressure vessel, coil stack assembly, latch assembly, and the drive rod assembly.

1. Pressure Vessel

   The pressure vessel includes a latch housing and a rod travel housing which are connected by a threaded, seal-welded maintenance joint which facilitates replacement of the latch assembly. The threaded connection forms the pressure boundary. The canopy seals are not pressure retaining and are designed to provide a means to control leaks through the threads. If leakage develops through a canopy seal weld, either a weld repair or a canopy seal clamp assembly can be used to repair or prevent the leak.

   The closure at the top of the rod travel housing is a threaded plug with a canopy seal weld for pressure integrity.

   The latch housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel and provides space for the drive rod during its upward movement as the control rods are withdrawn from the core.

2. Coil Stack Assembly

   The coil stack assembly includes the coil housings, an electrical conduit and connector, and three operating coils: (1) the stationary gripper coil, (2) the movable gripper coil, and (3) the lift coil.

   The coil stack assembly is a separate unit, which is installed on the drive mechanism by sliding it over the outside of the latch housing. It rests on the base of the latch housing without mechanical attachment.

Energizing the operating coils causes movement of the pole pieces and latches in the latch assembly.

3. Latch Assembly

The latch assembly includes the guide tube, stationary pole pieces, movable pole pieces, and two sets of latches: (1) the movable gripper latches and (2) the stationary gripper latches.

The latches engage grooves in the drive rod assembly. The movable gripper latches are moved up or down in inch steps by the lift pole to raise or lower the drive rod. The stationary gripper latches hold the drive rod assembly while the movable gripper latches are repositioned for the next inch step.

4. Drive Rod Assembly

The drive rod assembly includes a flexible coupling, a drive rod, a disconnect device, a disconnect rod, and a locking button.

The drive rod has inch grooves which receive the latches during holding or moving of the drive rod. The flexible coupling is attached to the drive rod and provides the means for coupling to the Rod Cluster Control Assembly.

The disconnect device, disconnect rod, and locking button provide positive locking of the coupling to the Rod Cluster Control Assembly and permit remote uncoupling of the drive rod from the Rod Cluster Control Assembly.

The control rod drive mechanism is a trip design. Tripping can occur during any part of the power cycler sequencing, if electrical power to the coils is interrupted.

The control rod drive mechanism is threaded and seal-welded on an adaptor housing on top of the reactor vessel, and is coupled to the Rod Cluster Control Assembly directly below. The mechanism is capable of raising or lowering a 360 pound load, (which includes the drive rod weight) at a rate of 45 inches/minute. Withdrawal of the Rod Cluster Control Assembly is accomplished by magnetic forces, while insertion is by gravity.

The mechanism internals are designed to operate in 650°F reactor coolant. The pressure vessel is designed to contain reactor coolant at 650°F and 2500 . The three operating coils are designed to operate at 392°F, with forced air cooling required to maintain that temperature as a normal maximum condition.

The control rod drive mechanism, shown schematically in Figure 3.9(N)-4, withdraws and inserts a Rod Cluster Control Assembly as shaped electrical pulses are received by the operating coils. An ON or OFF sequence, repeated by silicon-controlled rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is measured by 42 discrete coils mounted on the position indicator assembly surrounding the rod travel housing. Each coil magnetically senses the entry and presence of the top of the ferromagnetic drive rod assembly as it moves through the coil center line.

During plant operation the stationary gripper coil of the drive mechanism holds the Rod Cluster Control Assembly in a static position until a stepping sequence is initiated, at which time the movable gripper coil and lift coil are energized sequentially.

b. Rod Cluster Control Assembly Withdrawal

The Rod Cluster Control Assembly is withdrawn by repetition of the following sequence of events (refer to Figure 3.9(N)-4).

1. Movable Gripper Coil (B) - ON

The latch-locking plunger raises and swings the movable gripper latches into the drive rod assembly groove. A 0.047 inch axial clearance exists between the latch teeth and the drive rod.

2. Stationary Gripper Coil (A) - OFF

The force of gravity, acting upon the drive rod assembly and attached control rod, causes the stationary gripper latches and plunger to move downward 0.047 inch until the load of the drive rod assembly and attached control rod is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

3. Lift Coil (C) - ON

   The 5/8 inch gap between the movable gripper pole and the life pole closes and the drive rod assembly raises one step length (5/8 inch).

4. Stationary Gripper Coil (A) - ON

   The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 0.047 inch. The 0.047 inch vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

5. Movable Gripper Coil (B) - OFF

   The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

6. Lift Coil (C) - OFF

   The gap between the movable gripper pole and lift pole opens. The movable gripper latches drop 5/8 inch to a position adjacent to a drive rod assembly groove.

7. Repeat Step 1

   The sequence described above (Items 1 through 6) is termed as one step or one cycle. The Rod Cluster Control Assembly moves 5/8 inch for each step or cycle. The sequence is repeated at a rate of up to 72 steps per minute, and the drive rod assembly (which has a 5/8 inch groove) is raised 72 grooves per minute. The Rod Cluster Control Assembly is thus withdrawn at a rate up to 45 inches per minute.

c. Rod Cluster Control Assembly Insertion

The sequence for Rod Cluster Control Assembly insertion is similar to that for control rod withdrawal, except the timing of lift coil (C) ON and OFF is changed to permit lowering the control assembly.

1. Lift Coil (C) - ON

   The 5/8 inch gap between the movable gripper and lift pole closes. The movable gripper latches are raised to a position adjacent to a drive rod assembly groove.

2. Movable Gripper Coil (B) - ON

   The latch-locking plunger raises and swings the movable gripper latches into a drive rod assembly groove. A 0.047 inch axial clearance exists between the latch teeth and the drive rod assembly.

3. Stationary Gripper Coil (A) - OFF

   The force of gravity, acting upon the drive rod assembly and attached Rod Cluster Control Assembly, causes the stationary gripper latches and plunger to move downward 0.047 inch until the load of the drive rod assembly and attached Rod Cluster Control Assembly is transferred to the movable gripper latches. The plunger continues to move downward and swings the stationary gripper latches out of the drive rod assembly groove.

4. Lift Coil (C) - OFF

   The force of gravity and spring force separate the movable gripper pole from the lift pole and the drive rod assembly and attached rod cluster control drop down 5/8 inch.

5. Stationary Gripper (A) - ON

   The plunger raises and closes the gap below the stationary gripper pole. The three links, pinned to the plunger, swing the three stationary gripper latches into a drive rod assembly groove. The latches contact the drive rod assembly and lift it (and the attached control rod) 0.047 inch. The 0.047 inch vertical drive rod assembly movement transfers the drive rod assembly load from the movable gripper latches to the stationary gripper latches.

6. Movable Gripper Coil (B) - OFF

   The latch locking plunger separates from the movable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three movable gripper latches out of the drive rod assembly groove.

7. Repeat Step 1

   The sequence is repeated, as for Rod Cluster Control Assembly withdrawal, up to 72 times per minute which gives an insertion rate of 45 inches per minute.

d. Holding and Tripping of the Control Rods

During most of the plant operating time, the control rod drive mechanisms hold the Rod Cluster Control Assemblies withdrawn from the core in a static position. In the holding mode, only one coil, the stationary gripper coil (A), is energized on each mechanism. The drive rod assembly and attached Rod Cluster Control Assemblies hang suspended from the stationary gripper latches.

If power to the stationary gripper coil is cut off, the combined weight of the drive rod assembly and the Rod Cluster Control Assembly plus the stationary gripper return spring will push aside the latches and release the drive rod. The control rod falls by gravity into the core. The trip occurs as the magnetic field, holding the stationary gripper plunger against the stationary gripper fixed pole, collapses and the stationary gripper plunger is forced down by the stationary gripper return spring and weight acting upon the latches. After the Rod Cluster Control Assembly is released by the mechanism, it falls freely until the control rods enter the dashpot section of the thimble tubes in the fuel assembly.

3.9(N).4.2 Applicable CRDS Design Specifications

For those components in the Control Rod Drive System comprising portions of the reactor coolant pressure boundary, conformance with the General Design Criteria and 10 CFR 50, Section 50.55a, is discussed in Sections 3.1 and 5.2. Conformance with pertaining regulatory guides is discussed in Sections 4.5 and 1.8 and in Subsection 5.2.3.

a. Design Bases

Bases for temperature, stress on structural members, and material compatibility are imposed on the design of the reactivity control components.

b. Design Stresses

The CRDS is designed to withstand stresses originating from various operating conditions, as summarized in Table 3.9(N)-1.

Allowable Stresses: For normal operating conditions, Section III of the ASME Code is used. All pressure boundary components are analyzed as Class I components.

c. Dynamic Analysis

The cyclic stresses due to dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses on the CRDS.

d. Control Rod Drive Mechanisms

The Control Rod Drive Mechanism (CRDM) pressure housings are Class I components designed to meet the stress requirements for normal operating conditions of Section III of the ASME Boiler and Pressure Vessel Code. Both static and alternating stress intensities are considered. The stresses originating from the required design transients are included in the analysis.

A dynamic seismic analysis is required on the CRDMs when a seismic disturbance has been postulated, to confirm the ability of the pressure housing to meet ASME Code, Section III, allowable stresses and to evaluate the effect of the seismic event on the drop time.

The control rod drive mechanisms (CRDMs) are evaluated for the effects of postulated reactor vessel inlet branch line nozzle and outlet branch line nozzle breaks. A time-history analysis of the CRDMs is performed for the vessel motion discussed in Subsection 3.9(N).1.5. A model of the CRDMs is formulated with gaps at the upper CRDM supports modeled as nonlinear elements. The CRDMs are represented by beam elements with lumped masses. The translation and rotation of the vessel head is applied to this model. The resulting loads and stresses are compared to allowables to verify the adequacy of the system.

e. Control Rod Drive Mechanism Operational Requirements

The basic operational requirements for the CRDMs are:

1. 5/8 inch step
2. 147 inch travel
3. 360 pound maximum load
4. Step in or out at 45 inches/minute (72 steps/minute)
5. Electrical power interruption shall initiate release of drive rod assembly
6. Trip delay time of less than 150 milliseconds - free fall of drive rod assembly shall begin less than 150 milliseconds after power interruption, no matter what holding or stepping action is being executed with any load and coolant temperature of 100°F to 550°F
7. 40-year design life with normal refurbishment.

3.9(N).4.3 Design Loads, Stress Limits and Allowable Deformations

a. Pressure Vessel

The pressure retaining components are analyzed for loads corresponding to normal, upset, emergency and faulted conditions. The analysis performed depends on the mode of operation under consideration.

The scope of the analysis requires many different techniques and methods, both static and dynamic.

Some of the loads that are considered on each component where applicable are as follows:

1. Control rod trip (equivalent static load)
2. Differential pressure
3. Spring preloads
4. Coolant flow forces (static)
5. Temperature gradients
6. Differences in thermal expansion
    (a) Due to temperature differences
    (b) Due to expansion of different materials
7. Interference between components
8. Vibration (mechanically or hydraulically induced)
9. All operational transients listed in Table 3.9(N)-1
10. Pump overspeed
11. Seismic loads (Operating Basis Earthquake and Safe Shutdown Earthquake)
12. Blowdown forces (due to cold and hot leg branch line breaks).

The main objective of the analysis is to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Standard methods of strength of materials are used to establish the stresses and deflections of these components. The dynamic behavior of the reactivity control components has been studied using experimental test data and experience from operating reactors.

b. Drive Rod Assembly

All postulated failures of the drive rod assemblies either by fracture or uncoupling lead to a reduction in reactivity. If the drive rod assembly fractures at any elevation, that portion remaining coupled falls with, and is guided by the Rod Cluster Control Assembly. This always results in reactivity decrease for the control rods.

c. Latch Assembly and Coil Stack Assembly

1. Results of Dimensional and Tolerance Analysis

With respect to the CRDM system as a whole, critical clearances are present in the following areas:

Latch assembly (diametral clearances)

Latch arm-drive rod clearances

Coil stack assembly-thermal clearances

Coil fit in coil housing.

The following paragraphs define clearances that are designed to provide reliable operation in the control rod drive mechanism in these four critical areas. These clearances have been proven by life tests and actual field performance at operating plants.

(a) Latch Assembly - Thermal Clearances

The magnetic jack has several clearances where parts made of Type 410 stainless steel fit over parts made from Type 304 stainless steel. Differential thermal expansion is therefore important. Minimum clearances of these parts at 68°F is 0.011 inches. At the maximum design temperature of 650°F, minimum clearance is 0.0045 inches; at the maximum expected operating temperature of 550°F, minimum clearance is 0.0057 inches.

(b) Latch Arm - Drive Rod Clearances

The CRDM incorporates a load transfer action. The movable or stationary gripper latches are not under load during engagement, as previously explained, due to load transfer action.

Figure 3.9(N)-5 shows latch clearance variation with the drive rod as a result of minimum and maximum temperatures. Figure 3.9(N)-6 shows clearance variations over the design temperature range.

(c) Coil Stack Assembly - Thermal Clearances

The assembly clearances of the coil stack assembly over the latch housing were selected so that the assembly could be removed under all anticipated conditions of thermal expansion.

At 70°F, the inside diameter of the coil stack is 7.308/7.298 inches, and the outside diameter of the latch housing is 7.260/7.270 inches.

Thermal expansion of the mechanism, due to operating temperature of the CRDM, results in the minimum inside diameter of the coil stack being 7.310 inches at 222°F and the maximum latch housing outside diameter being 7.302 inches at 532°F.

Under the extreme tolerance conditions listed above, it is necessary to allow time for a 70°F coil housing to heat during a replacement operation.

To verify the acceptability of the above tolerances, four coil stack assemblies were removed from four hot control rod drive mechanisms, mounted on 11.035 inch centers on a 550°F test loop, allowed to cool, and then placed without incident.

(d) Coil Fit in Coil Housing

CRDM and coil housing clearances are selected so that coil heat up results in a close to tight fit. This is done to facilitate thermal transfer and coil cooling in a hot control rod drive mechanism.

3.9(N).4.4 Evaluation of Control Rod Drive Mechanisms and Supports

The Control Rod Drive Mechanisms (CRDMs) and CRDM support structure are evaluated for the loading combinations outlined in Table 3.9(N)-2.

A detailed finite element model of the CRDMs and CRDM supports is constructed using the WECAN computer program with beam, pipe, and spring elements. For the LOCA analysis, nonlinearities in the structure are represented. These include RPI plate impact, tie rods, and lifting leg clevis/RPV head interface. The time-history motion of the reactor vessel head, obtained from the RPV analysis is input to the dynamic model. Maximum forces and moments in the CRDMs and support structure are then determined. For the seismic analysis, the structural model is linearized and the floor response spectra corresponding to the CRDM tie rod elevation is applied to determine the maximum forces and moments in the structure.

The bending moments calculated for the CRDMs for the various loading conditions are compared with maximum allowable moments determined from a detailed finite element stress evaluation of the CRDMs. Adequacy of the CRDM support structure is verified by comparing the calculated stresses to the criteria given in ASME III, Subsection NF.

3.9(N).4.5 CRDS Performance Assurance Program

a. Evaluation of Material's Adequacy

The ability of the pressure housing components to perform throughout the design lifetime, as defined in the equipment specification, is confirmed by the stress analysis report required by the ASME Code, Section III.

Internal components subjected to wear will withstand a minimum of 3,000,000 steps without refurbishment, as confirmed by life tests (Reference 12). Latch assembly inspection is recommended after 2.5×10^6 steps have been accumulated on a single control rod drive mechanism.

To confirm the mechanical adequacy of the fuel assembly, the control rod drive mechanism and Rod Cluster Control Assembly, functional test programs have been conducted on a full-scale 12-foot control rod. The 12-foot prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 hours. The prototype mechanism accumulated about 3,000,000 steps and 600 trips. At the end of the test, the control rod drive mechanism was still operating satisfactorily. A correlation was developed to predict the amplitude of flow-excited vibration of individual fuel rods and fuel assemblies. Inspection of the drive line components did not reveal significant fretting.

These tests include verification that the trip time achieved by the CRDM meets the design requirement of 2.2 seconds from start of Rod Cluster Control Assembly motion to dashpot entry. This trip time requirement will be confirmed for each control rod drive mechanism prior to initial reactor operation and at periodic intervals after initial reactor operation, as required by the proposed Technical Specifications.

These tests have been reported in Reference 12.

In addition, dynamic testing programs have been conducted by Westinghouse and Westinghouse Licensees to demonstrate that control rod scram time is not adversely affected by postulated seismic events. Acceptable scram performance is assured by also including the effects of the allowable displacements of the driveline components in the evaluation of the test results.

There are no significant differences between the prototype control rod drive mechanisms and the production units. Design materials, tolerances and fabrication techniques are the same (see Section 4.5).

It is expected that all control rod drive mechanisms will meet specified operating requirements for the duration of plant life, with normal refurbishment. Latch assembly inspection is recommended after $12.5x10^6$ steps have been accumulated on a single CRDM.

If a Rod Cluster Control Assembly cannot be moved by its mechanism, shutdown margin is determined and, if required, adjustments in the boron concentration ensure that adequate shutdown would be achieved follow a trip. Thus, inability to move one Rod Cluster Control Assembly can be tolerated. More than one inoperable Rod Cluster Control Assembly could be tolerated, but would impose additional demands on the plant operator. Therefore, the number of inoperable Rod Cluster Control Assemblies has been limited to one as discussed in the proposed Technical Specifications.

In order to demonstrate proper operation of the control rod drive mechanism, and to ensure acceptable core power distributions during Rod Cluster Control Assembly partial-movement, checks are performed on the Rod Cluster Control Assemblies (refer to Technical Specifications). In addition, periodic drop tests of the Rod Cluster Control Assemblies are performed at each refueling shutdown, to demonstrate continued ability to meet trip time requirements. During these tests, the acceptable drop time of each assembly is not greater than 2.4 seconds, at full flow and operating temperature, from the beginning of motion to dashpot entry.

Actual experience in operating Westinghouse plants indicates excellent performance of control rod drive mechanisms.

All units are production tested prior to shipment, to confirm ability of the control rod mechanism to meet design specification operation requirements.

Each production CRDM undergoes a production test as listed below:

Test	Acceptance Criteria
Cold (ambient) hydrostatic	ASME Section III
Confirm step length and load transfer (stationary gripper	Step Length 5/8 ±0.015 inches axial movement
to movable gripper or movable gripper to stationary gripper)	Load Transfer 0.047 inches nominal axial movement
Cold (ambient) performance test at design load -	Operating Speed 45 inches/minute
5 full travel excursions	Trip Delay Free fall of drive rod to begin within 150 milliseconds

3.9(N).5 Reactor Pressure Vessel Internals

3.9(N).5.1 Design Arrangements

The reactor vessel internals are described as follows:

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and neutron shield pad assembly), the upper core support structure and the incore instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms, direct coolant flow past the fuel elements, direct coolant flow to the pressure vessel head, provide gamma and neutron shielding, and provide guides for the incore instrumentation. The coolant flows from the vessel inlet nozzles down the annulus between the core barrel and the vessel wall and then into a plenum at the bottom of the vessel. It then reverses and flows up through the core support and through the lower core plate. The lower core plate is sized to provide the desired inlet flow distribution to the core. After passing through the core, the coolant enters the region of the upper support structure and then flows radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles. A small portion of the coolant flows between the baffle plates and the core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum and exits through the vessel outlet nozzles.

a. Lower Core Support Structures

The major containment and support member of the reactor internals is the lower core support structure, shown in Figure 3.9(N)-7. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the neutron shield pads, and the core support which is welded to the core barrel. All the major material for this structure is Type 304 stainless steel. The lower core support structure is supported at its upper flange from a ledge in the reactor vessel head flange, and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are an axial baffle and a lower core plate, both of which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core support structure and principally the core barrel serve to provide passageways and control for the coolant flow. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a member through which the necessary flow distribution holes for each fuel assembly are machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the core support of the core barrel to provide stiffness and to transmit the core load to the core support. Adequate coolant distribution is obtained through the use of the lower core plate and core support.

The neutron shield pad assembly consists of four pads that are bolted and pinned to the outside of the core barrel. These pads are constructed of Type 304 stainless steel, and are approximately 48 inches wide by 148 inches long by 2.8 inches thick. The pads are located azimuthally to provide the required degree of vessel protection. Specimen guides in which material surveillance samples can be inserted and irradiated during reactor operation are attached to the pads. The samples are held in the guides by a preloaded spring device at the top and bottom to prevent sample movement. Additional details of the neutron shield pads and irradiation specimen holders are given in Reference 13.

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading, hydraulic loads and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell and partially through the lower support columns to the core support and, thence, through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell and distributed between the lower radial support to the vessel wall, and to the vessel flange. Transverse loads of the fuel assemblies are transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by upper core plate alignment pins which are welded into the core barrel.

The Main Radial Support System of the lower end of the core barrel is accomplished by "key" and "keyway" joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel clevis block is welded to the vessel inner diameter. Another Inconel insert block is bolted to each of these blocks and has a "keyway" geometry. Opposite each of these is a "key" which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity, and may be viewed as a beam supported at the top and bottom.

Radial and axial expansions of the core barrel are accommodated but transverse movement of the core barrel is restricted by this design. With this system, cyclic stresses in the internal structures are within ASME Code, Section III, limits. In the event of an abnormal downward vertical displacement of the internals following a hypothetical failure, energy-absorbing devices limit the displacement after contacting the vessel bottom head. The load is then transferred through the energy-absorbing devices of the internals to the vessel.

The energy absorbers, cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Assuming a downward vertical displacement the potential energy of the system is absorbed mostly by the strain energy of the energy-absorbing devices.

b. <u>Upper Core Support Assembly</u>

The upper core support assembly, shown in Figure 3.9(N)-8 and Figure 3.9(N)-9, consists of the top support plate assembly and the upper core plate, between which are contained support columns and guide tube assemblies. The support columns establish the spacing between the top support plate assembly and the upper core plate, and are fastened at top and bottom to these plates. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. The guide tube assemblies sheath and guide the control rod drive shafts and control rods. They are fastened to the top support plate, and are restrained by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the upper guide tube which is attached to the upper support plate and guide tube.

The upper core support assembly is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel which, in turn, engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 90 degrees from each other. Four slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies and control rods are thereby assured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly and is compressed by the reactor vessel head flange.

Vertical loads from weight, earthquake acceleration, hydraulic loads and fuel assembly preload are transmitted through the upper core plate via the support columns to the top support plate assembly and then the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the top support plate and upper core plate. The top support plate is particularly stiff to minimize deflection.

c. <u>Incore Instrumentation Support Structures</u>

The incore instrumentation support structure consists of a guide tubing system to convey and support flux thimbles penetrating the vessel through the bottom. Each detector assembly includes five fixed gamma/neutron flux detectors, and a thermocouple. (Figure 7.7-9 shows the Incore Detector Guide Tube configuration.)

The guide tubes extend from the seal table down through the concrete shield area and terminate in socket welds at the reactor vessel bottom head penetrations. The guide tube bend radius is 12 feet. The detector assemblies extend through the guide tubing and vessel penetrations, through hollow passages in the lower internals and finally through instrumentation support tubes in the fuel assemblies. The detector assemblies remain in place during operation but are pulled back approximately 13 feet at the seal table during refueling to avoid interference within the core. The detector assemblies are closed at the leading ends and sealed against the guide tubes at the seal table.

Mechanical seals between the retractable detector assemblies and guide tubes are provided at the seal table. During normal operation, the retractable detector assemblies are stationary and move only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal table is cleared for the retraction operation.

The incore instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence.

These are the only conditions which affect the incore instrumentation support structure.

3.9(N).5.2 Design Loading Conditions

The design loading conditions that provide the basis for the design of the reactor internals are:

a. Fuel assembly weight

b. Fuel assembly spring forces

c. Internals weight

d. Control rod trip (equivalent static load)

e. Differential pressure

f. Spring preloads

g. Coolant flow forces (static)

h. Temperature gradients

i. Differences in thermal expansion due to:

    1. Temperature differences

    2. Expansion of different materials

j. Interference between components

k. Vibration (mechanically or hydraulically induced)

l. One or more loops out of service

m. All operational transients listed in Table 3.9(N)-1

n. Pump overspeed

o. Seismic loads (Operating Basis Earthquake and Safe Shutdown Earthquake)

p. Blowdown forces (due to cold and hot leg branch line breaks).

The main objective of the design analysis is to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but to also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Both low and high cycle fatigue stresses are considered when the allowable amplitude of oscillation is established. Dynamic analysis on the reactor internals is provided in Subsection 3.9(N).2.

As part of the evaluation of design loading conditions, extensive testing and inspections are performed from the initial selection of raw materials up to and including component installation and plant operation. Among these tests and inspections, are those performed during component fabrication, plant construction, startup and checkout, and during plant operation.

3.9(N).5.3 Design Loading Categories

The combination of design loadings fit into either the normal, upset, emergency or faulted conditions, as defined in the ASME Code, Section III. However, it should be noted that by contract the reactor internals for the Seabrook plant preceed the applicability of Subsection NG of the ASME Code. Therefore, these internals are not "Code Stamped" and no specific stress report is required. Nevertheless, these reactor internals are designed to meet the intent of Subsection NG of the ASME Code.

Loads and deflections imposed on components due to shock and vibration are determined analytically and experimentally in both scaled models and operating reactors. The cyclic stresses due to these dynamic loads and deflections are combined with the stresses imposed by loads from component weights, hydraulic forces and thermal gradients for the determination of the total stresses of the internals.

The reactor internals are designed to withstand stresses originating from various operating conditions, as summarized in Table 3.9(N)-1.

The scope of the stress analysis problem is very large requiring many different techniques and methods, both static and dynamic. The analysis performed depends on the mode of operation under consideration.

3.9(N).5.4 Design Bases

The design bases for the mechanical design of the reactor vessel internals components are as follows:

a. The reactor internals in conjunction with the fuel assemblies direct reactor coolant through the core to achieve acceptable flow distribution and to restrict bypass flow so that the heat transfer performance requirements are met for all modes of operation. In addition, required cooling for the pressure vessel head is provided so that the temperature differences between the vessel flange and head do not result in leakage from the flange during reactor operation.

b. In addition to neutron shielding provided by the reactor coolant, a separate neutron pad assembly is provided to limit the exposure of the pressure vessel, in order to maintain the required ductility of the material for all modes of operation.

c. Provisions exist for installing incore instrumentation useful for the plant operation and vessel material test specimens required for a pressure vessel irradiation surveillance program.

d. The core internals are designed to withstand mechanical loads arising from the Operating Basis Earthquake, Safe Shutdown Earthquake and pipe ruptures, and meet the requirement of Item e. below.

e. The reactor has mechanical provisions which are sufficient to adequately support the core and internals, and to assure that the core is intact with acceptable heat transfer geometry following transients arising from abnormal operating conditions.

f. Following the design basis accident, the plant shall be capable of being shut down and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This implies that the deformation of certain critical reactor internals must be kept sufficiently small to allow core cooling.

The functional limitations for the core structures during the design basis accident are shown in Table 3.9(N)-12. To ensure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited to not exceed the value shown in Table 3.9(N)-12.

Details of the dynamic analyses, input forcing functions, and response loadings are presented in Subsection 3.9(N).2.

The basis for the design stress and deflection criteria is identified below:

a. Allowable Stresses

For normal operating conditions, Section III of the ASME Code is used as a basis for evaluating acceptability of calculated stresses. Both static and alternating stress intensities are considered.

It should be noted, that the allowable stresses in Section III of the ASME Code are based on unirradiated material properties. In view of the fact that irradiation increases the strength of the Type 304 stainless steel used for the internals, although decreasing its elongation, it is considered that use of the allowable stresses in Section III is appropriate and conservative for irradiated internal structures.

The allowable stress limits during the design basis accident used for the reactor internals are based on the draft of the 1971 edition of the ASME Code for Core Support Structures, Subsection NG, and the Criteria for Faulted Conditions.

b. Allowable Deflections

For normal operating conditions, downward vertical deflection of the lower core support plate is negligible.

For the loss-of-coolant accident plus the Safe Shutdown Earthquake condition, the deflection criteria of critical internal structures are the limiting values given in Table 3.9(N)-12. The corresponding no-loss-of-function limits are included in Table 3.9(N)-12 for comparison purposes with the allowed criteria.

The criteria for the core drop accident is based upon analyses which have to determine the total downward displacement of the internal structures following a hypothesized core drop resulting from loss of the normal core barrel supports. The initial clearance between the secondary core support structures and the reactor vessel lower head in the hot condition is approximately ½ inch. An additional displacement of approximately ¾ inch would occur due to strain of the energy absorbing devices of the secondary core support; thus the total drop distance is about 1¼ inches, which is insufficient to permit the tips of the Rod Cluster Control Assembly to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will never be used, except during a hypothetical accident of the core support (core barrel, barrel flange, etc.). There are 4 supports in each reactor. This device limits the fall of the core and absorbs much of the energy of the fall which otherwise would be imparted to the vessel. The energy of the fall is calculated assuming a complete and instantaneous failure of the primary core support and is absorbed during the plastic deformation of the controlled volume of stainless, loaded in tension. The maximum deformation of this austenitic stainless piece is limited to approximately 15 percent, after which a positive stop is provided to ensure support.

3.9(N).6 In-Service Testing of Pumps and Valves

Refer to Subsection 3.9(B).6.

3.9(N).7 References

1. "Documentation of Selected Westinghouse Structural Analysis Computer Codes," WCAP-8252, Revision 1, May 1977.

2. "Benchmark Problem Solutions Employed for Verification of the WECAN Computer Program," WCAP-8929, June 1977.

3. "Sample Analysis of a Class 1 Nuclear Piping System," prepared by ASME Working Group on Piping, ASME Publication, 1972.

4. Witt, F. J., Bamford, W. H., Esselman, T. C., "Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants during Postulated Seismic Events," WCAP-9283, March 1978.

5. Bogard, W. T., Esselman, T. C., "Combination of Safe Shutdown Earthquake and Loss-of-Coolant Accident Responses for Faulted Condition Evaluation of Nuclear Power Plants," WCAP-9279, March 1978.

6. "Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests," WCAP-8303-P-A (Proprietary) and WCAP-8317A (Nonproprietary), July 1975.

7. Bloyd, C. N. and Singleton, N. R., "UHI Plant Internals Vibration Measurement Program and Pre and Post Hot Functional Examinations," WCAP-8516-P (Proprietary) and WCAP-8517 (Nonproprietary), April 1975.

8. Bloyd, C. N., Ciaramitaro, W. and Singleton, N. R., "Verification of Neutron Pad and 17x17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant," WCAP-8766 (Proprietary) and WCAP-8780 (Nonproprietary), May 1976.

9. Takeuchi, K., et al., "MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal Hydraulic-Structure System Dynamics," WCAP-8708-P-A, Westinghouse Proprietary Class 2 / WCAP-8709 NES Class 3 (Non-Proprietary), September 1977.

10. Takeuchi, K., et al., "MULTIFLEX 3.0 A FORTRAN IV Computer Program for Analyzing Thermal Hydraulic-Structural System Dynamics Advanced Beam Model," WCAP-9735, Revision 2, Westinghouse Proprietary Class 2/ WCAP-9736, Revision 1, Non-Proprietary, February 1998.

11. Bohm, G. J. and LaFaille, J. P., "Reactor Internals Response Under a Blowdown Accident," Procedures, First Intl. Conf. on Structural Mech. in Reactor Technology, Berlin, September 20-24, 1971.

12. Cooper, F. W. Jr., "17x17 Drive Line Components Tests - Phase IB, II, III, D-Loop Drop and Deflection," WCAP-8446 (Proprietary) and WCAP-8449 (Nonproprietary), December 1974.

13. Kraus, S., "Neutron Shielding Pads," WCAP-7870, May 1972.

14. Final Rule Modifying 10 CFR 50 Appendix A, GDC-4 dated October 27, 1987 [52 FR 41288].

15. Letter, T. H. Essig (USNRC) to Lou Liberatori (WOG), "Safety Evaluation of Topical Report WCAP-15029, 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions', (TAC NO. MA1152)," November 10, 1998 (Enclosure 1 – Safety Evaluation Report).

16. Schwirian, R. E., et al., "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Load Conditions," WCAP-15029-P-A, Westinghouse Proprietary Class 2/WCAP-15030-NP-A, Revision 0, non-proprietary, January 1999.

3.10(B) SEISMIC QUALIFICATION OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

This section discusses the seismic qualification criteria, methods and procedures employed by the A-E for the qualification of seismic Category I instrumentation and electrical equipment within his scope of responsibility. It also covers the methods of analysis or testing of the supports for the electrical equipment and instrumentation. Seismic Category I instrumentation and electrical equipment are listed in Table 3.2-1 and Table 3.2-2. The corresponding discussion for the electrical equipment, instrumentation and supports provided by the NSSS supplier is found in Section 3.10(N).

3.10(B).1 Seismic Qualification Criteria

The criteria employed for seismic qualification of seismic Category I (Class 1E) electrical equipment and instrumentation, other than NSSS-related equipment, follow the guidelines recommended in IEEE Std. 344-1975 and Regulatory Guide 1.100.

The seismic Category I instrumentation and electrical equipment are designed to withstand, without loss of function or structural integrity, the combined effects of all normal operating loads and the seismic loads of the Safe Shutdown Earthquake (SSE) or the Operating Basis Earthquake (OBE), as defined in Subsection 3.7(B).1.

The seismic Category I instrumentation and electrical equipment are seismically qualified by using either analytical methods or testing, or by a combination of both, as follows:

a. The equipment which does not undergo a change of state is qualified by analysis or testing to establish structural adequacy. This includes cable tray systems, cable conduit systems and bus duct systems. When analytical methods are used to qualify the seismic Category I equipment or component, stresses resulting from the seismic acceleration effects are combined with stresses due to normal operating loads. Both horizontal and vertical seismic loads are assumed to occur simultaneously in the most unfavorable combinations, and the two horizontal and one vertical seismic load-induced stress components are combined by the square-root-of-the-sum-of-the-squares method. The stress levels due to the combined normal design loads and seismic loads are maintained within the stress levels set forth in appropriate design standards and codes. If there are no code requirements in the design of the equipment or portions thereof, then the stress level under the above combined loading, including the normal design loads plus SSE seismic load condition, is limited to 90 percent of the minimum yield strength for the material. In addition, the equipment or component design is reviewed to assure that any resulting deflections or distortions do not prevent the proper functioning of the equipment, nor endanger adjacent equipment or components. The criteria for selecting a static load analysis or dynamic modal analysis are decided by the complexity of the equipment, as delineated in IEEE Std. 344-1975.

   When testing methods are used to qualify cable tray systems, the testing performance will meet the intent of IEEE Std. 344-1975.

b. The equipment or components which must be capable of undergoing a change of state are qualified by a combination of seismic and functional testing both during and after an earthquake of magnitude up to and including the SSE. The equipment or components are tested for ability to retain structural integrity, and no malfunction is permitted where such a malfunction could jeopardize the capability to safely shut down the reactor and/or mitigate offsite exposures. When testing is used to qualify seismic Category I equipment or components, testing is performed in accordance with IEEE Std. 344-1975.

c. Seismic Category I equipment supports, including cabinets, panels, consoles and instrumentation racks are qualified by either analysis or testing using appropriate horizontal and vertical floor response spectra at the building and elevation at which they are installed. When analysis is used to qualify the equipment supports, the stress criteria are in accordance with AISC Manual of Steel Construction.

d. Seismic qualification tests are conducted for battery prototypes with cables (or equivalent batteries with cables) in accordance with the requirements of IEEE 323-1974 and IEEE 344-1975. The battery racks are qualified by appropriate seismic analyses which include the battery masses. The Class 1E transformers are seismically qualified by test using an appropriate test apparatus which includes the supporting structures or cabinets and all relevant appurtenances including cooling accessories. This effort is included in the Class 1E Unit Substation Qualification as noted in Table 3.10(B)-1.

3.10(B).2 <u>Methods and Procedures for Qualifying Instrumentation and Electrical Equipment</u>

Seismic Category I instrumentation and electrical equipment (other than NSSS) were qualified either by analysis, by testing, or by a combination of testing and analysis as indicated in Table 3.10(B)-1, to confirm their functional operability during and after an earthquake up to and including the SSE.

All specifications for seismic Category I instrumentation and electrical equipment included the seismic design criteria are discussed in Subsection 3.10(B).1 as a part of the design condition. Tests or analyses are performed in accordance with the criteria and the intent of IEEE Std. 344-1975 and Regulatory Guide 1.100. Amplified floor response spectra applicable to the particular equipment locations were included as a part of the specification. The seismic intensities indicated in these spectra were used by the manufacturer for seismic qualification of the equipment and capability documentation.

Certified test report and/or analytical calculations were obtained from each vendor to confirm that the purchased seismic Category I instrumentation and electrical equipment will perform its function when subjected to the stipulated seismic loading conditions of the SSE. All test reports and calculations were certified by a registered professional engineer, skilled in the applicable specialty, and by a responsible officer of the manufacturer or vendor.

Equipment anchor loadings and details, such as size and spacing of anchor bolts or welds, were obtained from the equipment manufacturers for use in designing foundations or supporting floors for compatibility with the seismic anchor loading of the equipment.

3.10(B).3 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

3.10(B).3.1 Electrical Equipment and Instrument Supports

The qualification of supports for main control boards, cabinets, panels and instrument racks, as well as supports for electrical equipment such as battery racks, was accomplished using one of the methods discussed in Subsection 3.10(B).1. The methods used in evaluating the supports were tested under simulated conditions and analytical approaches. Analytical methods were employed to anchor the supports. Amplified floor response spectra for the locations where the equipment is mounted are provided to the equipment supplier who is responsible for qualifying the equipment. Supports for instruments and electric equipment are attached by bolting or welding to anchor plates fabricated of ASTM A36 steel, either embedded in the concrete with stud anchors or surface-mounted to the concrete using bolt anchors. In either case, they were designed to prevent uplift or overturning effects due to seismic forces.

3.10(B).3.2 Cable Tray Supports

Raceway systems when used to carry safety-related circuit cables, are designed or tested to withstand the seismic forces which would be experienced during an SSE due to the weight of the cables, raceways and supports. Cable tray load-deflection curves were used to formulate a simplified analytical model of the tray which was then coupled to the analytical model of the supports. Load deflection parameters may be established by testing to formulate the analytical models of the cable tray supports. The response spectra method was used to analyze the overall analytical model and to design the support structures, while complying with the tray support system functional requirement.

The Cable Tray Support System was analyzed for dead load combined with the OBE loads, with the stress criteria based on the allowable stresses of the AISC Specification on structural steel for buildings and the engineering information for strut members published by Unistrut, Powerstrut or Superstrut. For dead loads combined with SSE loads, the stresses are limited to 90 percent of yield stress for the material involved. For the seismic loads, the actual natural frequency response of each support system was calculated and the appropriate seismic acceleration factor was selected from the amplified floor response spectra (see Section 3.7).

The cable tray supports consist of structural shapes and strut members. The structural steel portion of the supports is fabricated from ASTM A36 steel. Each support is connected to either the structural steel framing, the concrete floor slab or the concrete wall of the building structures.

3.10(B).4 Operating License Review

Results obtained from the seismic qualification tests and/or analyses performed for the various categories of equipment listed in Table 3.10(B)-1 are documented in the reference documents (see contract no.) listed in the same table. These documents are maintained in the Information Management System files, and in UE&C Foreign Print files as required by ANSI N45.2.9.

United Engineers and Constructors has reviewed these documents as part of the routine vendor document review process to assure that the seismic qualification methods are acceptable, and the results thereof sufficiently demonstrate the qualification of the equipment.

3.10(N) SEISMIC QUALIFICATION OF CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

This section presents information to demonstrate that instrumentation and electrical equipment classified as seismic Category I is capable of performing designated safety-related functions in the event of a safe shutdown earthquake (SSE). The information presented includes identification of the seismic Category I instrumentation and electrical equipment that are within the scope of the Westinghouse Nuclear Steam Supply System (NSSS), the seismic qualification criteria employed and, for each item of equipment, the designated safety-related functional requirements, definition of the applicable seismic environment and documentation of the qualification process employed to demonstrate the required seismic capability.

3.10(N).1 Seismic Qualification Criteria

3.10(N).1.1 Qualification Standards

The methods of meeting the general requirements for seismic qualification of Category I instrumentation and electrical equipment, as described by General Design Criteria (GDC) 1, 2 and 23, are described in Section 3.1. The general methods of implementing the requirements of Appendix B to 10 CFR Part 50 are described in Chapter 17.

The Commission's recommendations concerning the methods to be employed for seismic qualification of electrical equipment are contained in Regulatory Guide 1.100, which endorses IEEE 344-1975. Westinghouse meets this standard, as modified by Regulatory Guide 1.100, by either type test, analysis, or an appropriate combination of these methods. Westinghouse will meet this commitment employing the methodology described in the final staff approved version of Reference 1.

3.10(N).1.2 Performance Requirements for Seismic Qualification

Reference 2 contains an equipment qualification data package (EQDP) for every item of instrumentation and electrical equipment classified as seismic Category I within the Westinghouse NSSS scope of supply. Table 3.10(N)-1 identifies the seismic Category I equipment supplied by Westinghouse for this application and references the applicable EQDP contained in Supplement 1. Each EQDP in Supplement 1 contains a section entitled "Performance Specification." This specification establishes the safety-related functional requirements of the equipment to be demonstrated during and after a seismic event. The required response spectrum (RRS) employed by Westinghouse for generic seismic qualification is also identified in the specification, as applicable. The spectra employed have been selected to envelope the plant specific spectra defined in Section 3.7.

3.10(N).1.3 Acceptance Criteria

Seismic qualification must demonstrate that seismic Category I instrumentation and electrical equipment is capable of performing designated safety-related functions during and after an earthquake of magnitude up to and including the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) without the initiation of undesired spurious actuation which might result in consequences adverse to safety. The qualification must also demonstrate the structural integrity of mechanical supports and structures at the OBE level. Some permanent mechanical deformation of supports and structures is acceptable at the SSE level providing that the ability to perform the designated safety-related functions is not impaired.

3.10(N).2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

In accordance with IEEE 344-1975, seismic qualification of safety-related electrical equipment is demonstrated by either type testing, analysis or a combination of these methods. The choice of qualification method employed by Westinghouse for a particular item of equipment is based upon many factors including: practicability, complexity of equipment, economics, availability of previous seismic qualification to earlier standards, etc. The qualification method employed for a particular item of equipment is identified in the individual Equipment Qualification Data Packages (EQDPs) of Reference 2.

3.10(N).2.1 Seismic Qualification by Type Test

From 1969 to mid-1974 Westinghouse seismic test procedures employed single axis sine beat inputs in accordance with IEEE 344-1971 to seismically qualify equipment. The input form selected by Westinghouse was chosen following an investigation of building responses to seismic events as reported in Reference 3. In addition, Westinghouse has conducted seismic retesting of certain items of equipment as part of the Supplemental Qualification Program (Reference 4). This retesting was performed at the request of the NRC staff on agreed selected items of equipment employing multi-frequency, multi-axis test inputs (Reference 5), to demonstrate the conservatism of the original sine beat test method with respect to the modified methods of testing for complex equipment recommended by IEEE 344-1975.

The original single axis sine beat testing and the additional retesting completed under the Supplemental Test Program has been the subject of generic review by the Staff. For equipment which has been previously qualified by the single axis sine beat method and included in the NRC seismic audit and, where required by the Staff, the Supplemental Qualification Program (Reference 4), no additional qualification testing is required to demonstrate acceptability to IEEE 344-1975 provided that:

a. The Westinghouse aging evaluation program for aging effects on complex electronic equipment located outside containment demonstrates there are no deleterious aging phenomena. In the event that the aging evaluation program identifies materials that are marginal, either the materials will be replaced or the projected qualified life will be adjusted.

b. Any changes made to the equipment due to a. above or due to design modifications do not significantly affect the seismic characteristics of the equipment.

c. The previously employed test inputs can be shown to be conservative with respect to applicable plant specific response spectra.

This equipment is identified in Reference 1, Table 7.1 and the test results in the applicable EQDPs of Reference 2.

For equipment tests after July 1974 (i.e., new designs, equipment not previously qualified or previously qualified equipment that does not meet a, b, and c above), seismic qualification by test is performed in accordance with IEEE 344-1975. Where testing is used, multi-frequency multi-axis inputs are developed by the general procedures outlined in Reference 5. The test results contained in the individual EQDPs of Reference 2 demonstrate that the measured Test Response Spectrum envelopes the applicable Required Response Spectrum (RRS) defined for generic testing, as specified in Section I of the EODP (Reference 2). Qualification for plant-specific use is established by verification that the generic RRS specified by Westinghouse envelopes the applicable plant specific response spectrum. Alternative test methods such as single frequency and single axis inputs are used in selected cases as permitted by IEEE 344-1975 and Regulatory Guide 1.100.

3.10(N).2.2 Seismic Qualification by Analysis

The structural integrity of safety-related motors (Table 3.10(N)-1, EQDP-AE-2 and 3) is demonstrated by a static seismic analysis in accordance with IEEE 344-1975, with justification. Should analysis fail to show the resonant frequency to be significantly greater than 33 Hz, a test is performed to establish the motor resonant frequency. Motor operability during a seismic event is demonstrated by calculating critical deflections, loads and stresses under various combinations of seismic, gravitational and operational loads. The worst case (maximum) values calculated are tabulated against the allowable values. On combining these stresses, the most unfavorable possibilities are considered in the following areas: (1) maximum rotor deflection, (2) maximum shaft stresses, (3) maximum bearing load and shaft slope at the bearings, (4) maximum stresses in the stator core welds, (5) maximum stresses in the stator core to frame welds, (6) maximum stresses in the motor mounting bolts and (7) maximum stresses in the motor feet.

The analytical models employed and the results of the analysis are described in Section 4 of the applicable EQDPs (Reference 2).

3.10(N).3 Method and Procedures for Qualifying Supports of Electrical Equipment and Instrumentation

Where supports for the electrical equipment and instrumentation are within the Westinghouse NSSS scope of supply, the seismic qualification tests and/or analysis are conducted including the supplied supports. The EQDPs contained in Reference 2 identify the equipment mounting employed for qualification purposes and establish interface requirements for the equipment to ensure that subsequent in-plant installation does not prejudice the qualification established by Westinghouse.

3.10(N).4 Operating License Review

The results of tests and analyses that ensure that the criteria established in Subsection 3.10(N).1 have been satisfied employing the qualification methods described in Subsections 3.10(N).2 and 3.10(N).3 are included in the individual EQDPs contained in Reference 2 as they become available.

3.10(N).5 References

1. Butterworth, G. and Miller, R.B., "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," WCAP-8587, Revision 2, February 1979.

2. "Equipment Qualification Data Packages," Supplement I to WCAP-8587, November 1978.

3. Morrone, A., "Seismic Vibration Testing with Sine Beats," WCAP-7558, October 1971.

4. NS-CE-692, Letter dated July 10, 1975 from C. Eicheldinger (Westinghouse) to D.B. Vasello (NRC).

5. Jarecki, S.J., "General Method of Developing Multi-Frequency Biaxial Test Inputs for Bistables," WCAP-8624 (Proprietary) September 1975 and WCAP-8695 (Nonproprietary) August 1975.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

Safety-related equipment must be capable of maintaining functional operability under conditions postulated to occur during its installed life. This requirement is embodied in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants," Criteria 1, 2, 4, 23 and 50.

The NRC has used a variety of methods to ensure that these general requirements are met for electric equipment important to safety. For nuclear plants after 1971, qualification was judged on the basis of IEEE Std 323-1971, "Trial Use Standard - General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations." For plants whose construction permit Safety Evaluation Reports (CP-SER) were issued after July 1, 1974, the Commission has used Regulatory Guide 1.89 which endorses IEEE Std 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," subject to supplementary provisions.

The NRC subsequently issued more definitive criteria in NUREG-0588, "Interim Staff Position on Environmental Qualification (EQ) of Safety-Related Electrical Equipment," which contains two sets of criteria:

a. Category I, for plants whose CP-SER was issued after July 1, 1974, incorporates and supplements IEEE Std 323-1974

b. Category II, for plants whose CP SER was issued before July 1, 1974, incorporates and supplements IEEE Std 323-1971.

Because the Seabrook CP-SER was issued August 14, 1974, NUREG-0588, Category I Criteria are applicable.

Environmental qualification of safety-related equipment located in a mild environment is ensured by conformance to the general quality and surveillance requirements identified in 10 CFR, Part 50, Appendix B, and Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," as implemented in Chapter 17. A separate report, "Environmental Qualification of Electrical Equipment Important to Safety" (E.Q. Report), has been developed which provides the detailed information required to certify qualification of electrical components located in a potential harsh environment (Reference 1). The report describes the scope of the Seabrook E.Q. Program, including the station operations and maintenance E.Q. program. The E.Q. program development, methodology and the equipment qualification status is also discussed in the reports.

The Mechanical Equipment Qualification status was submitted January 1984 (Reference 2). A revised final Mechanical Equipment Qualification (MEQ) report was submitted in April 1986.

The NSSS-supplied equipment and BOP-supplied equipment have been evaluated for the Seabrook plant specific environments by the same E.Q. programs. The E.Q. Reports and MEQ Reports describe the methodology and results for all plant equipment.

3.11.1 Determination of Equipment to be Environmentally Qualified

The list of systems and equipment was established by considering those components required to safely shut down and mitigate the following accidents:

a. Loss-of-Coolant Accident (LOCA)

b. Main Steam Line Break (MSLB)

c. Fuel Handling Accident (FHA)

d. High Energy Line Break (HELB)

e. Moderate Energy Line Break (MELB).

Specifically included according to NRC guidance are those systems required to achieve or support the following:

a. Emergency Reactor Shutdown

b. Containment Isolation

c. Reactor Core Cooling

d. Containment Heat Removal

e. Core Residual Heat Removal

f. Prevention of Radioactive Releases to the Environment.

Section 2.0 of the E.Q. Report describes the scope and criteria for listing Electrical Equipment on the Harsh Environment Equipment List. Appendix A of the E.Q. Report contains the actual list of equipment. The list of mechanical equipment required to operate during and following an accident is in Appendix A of a report entitled "Environmental Qualification of Mechanical Equipment," submitted via Reference 8.

3.11.2 Establishment of Environmental Service Conditions

Environmental conditions have been determined for normal and accident conditions. The specific normal, abnormal, accident and post-accident environments for Seabrook Station have been calculated for various environmental zones of all safety-related buildings. The environmental parameters of interest are temperature, pressure, humidity, radiation, chemical spray and submergence. Section 3.0 of the E.Q. Report describes in detail the methodology and assumptions used in calculating the various building zone environments. The maximum, normal, and minimum environmental parameters are provided in Updated FSAR Figure 3.11-1 (Figure 3-1, "Service Environment Chart," of the E.Q. Report). Table 3-2 of the E.Q. Report is a summary of flood levels in the safety-related buildings which could result from postulated accidents.

The accident radiation doses have been evaluated up to an average core burnup of 45,000 MWD/MTU and an analyzed core power of 3659 MWT.

The plant HVAC systems were designed to maintain the temperature and humidity within the normal/accident limits defined in the Service Environment Chart. The HVAC system design is described in detail in Section 9.4. Most safety-related equipment necessary to perform required safe shutdown and accident mitigation functions is located in areas where the environment is controlled by redundant safety-related ventilation systems. This assures that the environment and therefore the safety-related equipment located in these areas will not be adversely affected by a ventilation equipment failure. The environmental parameters for areas where redundant, safety-related ventilation systems are not provided reflect an assumed loss of ventilation in the Service Environment Chart. This assures that safety-related equipment located in these areas is qualified to function properly in an environment resulting from a loss of ventilation.

3.11.2.1 Main Steam Line Break (MSLB) Effect on Environmental Qualification of Equipment

Environmental conditions due to certain size MSLBs postulated both inside and outside of containment cause superheated temperatures in the immediate environmental zone. In June 1984, Westinghouse issued a letter/notice of a possible unreviewed safety question (Reference 3). Later in December 1984, NRC issued IE Information Notice No. 84-90 regarding the same subject for plant review and applicability (Reference 4).

Seabrook Station has reviewed these conditions and has determined that the environmental parameters postulated for MSLB inside containment are not exceeded. The result of Westinghouse superheat sensitivity analyses presented in WCAP-8822, Supplement 2, shows that the addition of superheat has a negligible impact on containment pressure and temperature for atmospheric and sub-atmospheric containment designs. This result is due primarily to the following characteristics:

a. Containment peak pressure and temperature are predominately affected by the total mass and energy release to the containment

b. The addition of superheat has no impact on steamline break mass releases and minimal impact on energy release.

Westinghouse has further stated that the above characteristics are independent of the steam generator type. Although the results and conclusions of WCAP-8822, Supplement 2, are based on a Model 51 steam generator, they are applicable to the Seabrook Model F steam generator.

The review of MSLB outside of containment for Seabrook indicates the main steam and feedwater pipe chases are the only affected building locations.

A closer detailed review of the MS&FW pipe chase design at Seabrook has shown that Seabrook can achieve a safe shutdown under any postulated superheated temperature profile due to a MSLB. This is achieved principally by the separation criteria conceptually designed into these building areas. Seabrook has two separated MS&FW pipe chase areas exiting the east and west sides of containment. Each pipe chase houses the feedwater and main steam piping for two of the four steam generators. The piping is designed under the concepts of "superpipe" (i.e., low stress allowables and upgraded ISI Program). Since the Westinghouse requirement is for a minimum two steam generator cooldown, Seabrook could safely shutdown under a postulated MSLB in the MS&FW pipe chase designed with "superpipe," using the alternate pipe chase. The safe shutdown is described in detail in the Seabrook Station Appendix R report (Reference 5). The MS&FW pipe chase houses the MS&FW containment isolation valves, main steam safety valves, atmospheric dump valves and MS supply valves to the emergency feed pump turbine. This equipment has been Environmentally Qualified to perform its design basis function during a postulated MSLB outside Containment.

3.11.3 Qualification Testing and Analysis of Electrical Equipment

Electrical equipment was evaluated to ensure that it will function as required when exposed to normal and postulated accident environments. All testing and analysis conformed to the requirements of IEEE 323-1974 and the guidance of NUREG-0588, Category I. The various testing parameters to demonstrate qualification of equipment include: functional criteria, test sequence, aging methodology, accident environment, margins, electrical connection interface. These parameters and their application to the Seabrook qualification testing program are described to a greater extent in Section 4.0 of the E.Q. Report.

The NSSS-supplied components were tested by a generic equipment qualification program conducted by Westinghouse. This program was described by Westinghouse WCAP-8587, Revision 2, and has been reviewed and accepted by the NRC (Reference 6). The Westinghouse program resulted in generic environmental qualification data packages (EQDP) for NSSS electrical equipment. These have been issued as Supplement 1 to WCAP-8587 (Reference 7).

3.11.4 Methodology for Evaluating Environmental Qualification to Plant Service Conditions

A comparison of environmental qualification test results to equipment service conditions in the Seabrook plant was completed for all equipment located in harsh environmental zones. This engineering analysis results in a documentation file, called an Environmental Qualification File (EQF), which certifies qualification to the plant environments. A description of the scope and content of a typical EQF is detailed in Section 4.2 of the E.Q. Report. Each Westinghouse generic EQDP is evaluated as any other vendor test report, and a corresponding Seabrook EQF is established to verify site specific qualification.

In addition, a verification program has been developed and completed to ensure that the vendor equipment tested was either identical or similar to the installed plant equipment.

The results of the Seabrook Qualification Analysis are described in Appendix B "Qualification Evaluation Worksheets" of the E.Q. Report.

3.11.5 Operational Phase and Maintenance of Environmental Qualification Program

The purpose of the Seabrook Station Equipment Qualification Program is the preservation of the qualification of equipment that is important to safety. In order to accomplish the task, the station has developed Design Control, Procurement and Maintenance Programs. Each program has incorporated the requirements of environmental qualification according to the functional requirements of the program. The station programs and procedures are prepared to maintain proper design control for plant modifications, procurement of new equipment and spare parts. The station maintenance program is designed to provide preventative as well as corrective maintenance which is identified by field operational experience. Section 5.0 of the E.Q. Report provides a more detailed discussion of the operational and maintenance programs.

3.11.6 References

1. PSNH letter SBN-886, "Equipment Qualification of Electrical Equipment; SER Outstanding Issue No. 6," dated October 31, 1985, from John DeVincentis to George W. Knighton.

2. PSNH letter SBN-608, "Mechanical Equipment Qualification," dated January 5, 1984, from John DeVincentis to George W. Knighton.

3. Westinghouse letter NAH-2495, "Environmental Qualification of Equipment for High Energy Line Breaks Outside of Containment," dated June 11, 1984, from D.P. Adomaitis to R. J. DeLoach.

4. NRC IE Information Notice No. 84-90, "Main Steam Line Break Effect on Environmental Qualification of Equipment," dated December 7, 1984.

5. PSNH letter SBN-904, "Fire Protection of Safe Shutdown Capability," dated December 2, 1985, from John DeVincentis to George W. Knighton.

6. Butterworth, G. and Miller, R.B., "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," WCAP-8587, Revision 2, February 1979.

7. "Equipment Qualification Data Packages," Supplement 1 to WCAP-8587, November 15, 1978.

8. PSNH letter SBN-1005, "Mechanical Equipment Qualification," dated April 16, 1986, from John DeVincentis to Vincent N. Noonan.

9. PSNH letter SBN-988, "Environmental Qualification; Post Accident Operability Time," dated April 3, 1986, from John DeVincentis to Vincent S. Noonan.

10. PSNH SBN-998, "Response to Environmental Qualification Audit Observation," dated April 10, 1986, from John DeVincentis to Vincent S. Noonan.

11. PSNH letter SBN-1031, "Environmental Qualification, "Evaluation Worksheets," dated May 7, 1986, from John DeVincentis to Vincent S. Noonan.

SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 3
DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

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TABLE 3.2-1 SEISMIC CATEGORY I STRUCTURES, SYSTEMS AND COMPONENTS

A.	**SEISMIC CATEGORY I STRUCTURES SYSTEM AND COMPONENTS (See Note 4)**	**UFSAR Reference Section**
1.	<u>Containment Structure</u> Cylinder Dome Base Mat Liner Plate	3.8.1
2.	Containment Internal Structures, including Fill Mat and Emergency Sump and Debris Strainer, and Debris Interceptors	3.8.3
3.	<u>Other Seismic Category I Structures</u> Containment Enclosure Building Containment Equipment Hatch Missile Shield Containment Enclosure Ventilation Area Control and Diesel Generator Building Control Room Makeup Air Intake Structures Emergency Feedwater Pump Building, including Electrical Cable Tunnels and Penetration Areas (Control Building to Containment) Enclosure for Condensate Storage Tank Fuel Storage Building Main Steam and Feedwater Pipe Chase (East), including East Penetration Area Main Steam and Feedwater Pipe Chase (West), including Mechanical Penetration Area and Personnel Hatch Area Piping Tunnels Pre-Action Valve Building Primary Auxiliary Building, including Residual Heat Removal (RHR) Equipment Vault Safety-Related Electrical Duct Banks and Manholes Service Water Access Vault Service Water Cooling Tower, including Switchgear Rooms Service Water Pumphouse Tank Farm (Tunnels), including Dikes and Foundations for Refueling Water Storage Tank (RWST) and Reactor Makeup Water Storage Tank Waste Processing Building Circulating Water Pumphouse Concrete below El 21'-0	3.8.4
4.	Foundations for Seismic Category I Structures	3.8.5

B.	SEISMIC CATEGORY I MECHANICAL EQUIPMENT AND COMPONENTS SYSTEM AND COMPONENTS	UFSAR Reference Section
1.	<u>Containment Equipment</u> Equipment Hatch Personnel Lock Mechanical Penetrations Electrical Penetrations	3.8.2
2.	ASME Code Class 1, 2 and 3 Component Supports and Core Support Structures	3.9.3
3.	<u>Control Rod Drive System</u> Control Rod Drive System CRDM Seismic Support Platform (see Note 1) CRDM Seismic Support Space Plates (see Note 1) CRDM Seismic Support Tie Rod Assemblies (see Note 1)	3.9.4
4.	Reactor Pressure Vessel Internals	3.9.5
5.	<u>Reactor Vessel</u> Reactor Vessel Head Studs, Nuts and Washers Reactor Vessel Shoes and Shims Irradiation Sample Holder (see Note 2)	5.3
6.	<u>Incore Instrumentation</u> Seal Table Assembly Guide Tubing (see Note 3))	7.7.1.9
7.	New Fuel Storage Racks	9.1.1
8.	<u>Spent Fuel Storage</u> Spent Fuel Pool Spent Fuel Storage Racks	9.1.2
9.	<u>Fuel Handling System</u> Reactor Vessel Head and Upper Internals Lifting Device (portions that furnish support to CFDMs) Spent Fuel Handling Tool Fuel Transfer System Fuel Transfer Tube and Flange Fuel Transfer Tube Outer Sleeve Expansion Joints Cask Handling Crane Trolley Frame and Main Hoist Machinery required to hold the load.	9.1.4

C. SEISMIC CATEGORY I FLUID SYSTEMS AND COMPONENTS SYSTEM AND COMPONENTS	UFSAR Reference Section
Seismic Category I Fluid Systems and Components are identified in Table 3.2-2	

D. SEISMIC CATEGORY I ELECTRICAL SYSTEMS AND COMPONENTS	UFSAR Reference Section
1. <u>Instrumentation and Controls</u>	7.0
Reactor Trip Switchgear	
Process Control Equipment Cabinets	
Westinghouse Solid-State Protection System and Safeguard Test Cabinets	
Nuclear Instrumentation System Cabinets	
Excore Neutron Detectors (Power Range and Wide Range)	
Pressure Transmitters	
Differential Pressure Transmitters	
Resistance Temperature Detectors	
Thermocouples	
Main Control Boards	
Remote Shutdown Panels	
Safety-Related Pilot Solenoid Valves	
Safety-Related Externally Mounted Limit Switches	
Instruments and Controls for use in Seismic Category I systems and components which are required to perform nuclear safety-related functions	
Instrumentation Supports, Fittings and Accessories	
Hydrogen Recombiner Package	6.2.5
Airborne and Particulate Radioactivity Monitors (APRM)	12.3.4
Online Purge Monitor	
Containment Atmosphere Monitor	
Control Room Air Intake Monitor	
Area Radiation Monitors	12.3.4
Containment Area Monitor	
Refueling Canal Monitor (mounted on Manipulator Crane)	
Accident Monitoring Instrumentation (Design Category I)	7.5
Seismic Monitoring Instrumentation	
Reactor Vessel Level Indication System	
High Energy Line Break Thermocouples	
Containment Building Level Transmitters	
Main Steam Isolation Valve Logic System	
Post Accident Sampling Isolation Valve Control Panel	
Remote Safe Shutdown Disabling Control Panel	
Radiation Monitoring Isolation Valves Control Panels	
Inadequate Core Cooling Monitor	4.4.6.5.7.5

D.	**SEISMIC CATEGORY I ELECTRICAL SYSTEMS AND COMPONENTS (Continued)**	**UFSAR Reference Section**
2.	Onsite Power Systems 4160V Switchgear (ESF Buses) 4160V Nonsegregated bus duct between ESF buses and diesel generators 4000V and 460V Motors (associated with ESF) Diesel Generators including those auxiliaries necessary for operation, e.g. governor, voltage regulator and excitation system. (See Note 6) Diesel Generators Control Panels 480V Motor Control Centers (associated with ESF) 480V Unit Substations (ESF buses) 4160V to 480V Transformers (associated with ESF) 120V Vital Panel Boards Containment Penetration Assemblies including primary and backup fault current protective devices. (See Note 7) Power Cables (5 kV and 600V) Instrumentation and Control Cables associated with ESF, including underground cable systems, e.g., cables in duct banks and cable splices. Emergency Power Sequencing System Conduit and Cable Tray Raceway System (nuclear safety-related) (See Note 5) 125V Batteries and battery racks (nuclear safety-related) Battery Chargers (nuclear safety-related) 125V DC Switchgear (nuclear safety-related) 125V DC Panelboards (nuclear safety-related) Inverters (vital instrument buses) Electrical Supports, Fittings and Accessories (i.e. connectors, terminal blocks, etc.) nuclear safety-related)	8.3.1, 8.3.2

E.	SEISMIC CATEGORY I HVAC SYSTEMS AND COMPONENTS	UFSAR Reference Section
1.	Control Building Control Room Air Conditioning System Remote Air Intake Ventilation System Cable Spreading Room Ventilation System Battery Room Ventilation System Switchgear Room Ventilation System	
2	Diesel Generator Building Diesel Generator Room Ventilation System	
3.	Fuel Storage Building Fuel Storage Building Ventilation System	
4.	Primary Auxiliary Building Equipment Vault Area Ventilation System Containment Enclosure Area Ventilation System	
5.	Service Water Pumphouse Service Water Pumphouse Building Ventilation System	
6.	Miscellaneous Areas Main Steam and Feed Water Pipe Chases (East and West) Ventilation System Mechanical Penetration Area Ventilation System Emergency Feed Water Pumphouse Building Ventilation System Cooling Tower Ventilation System	

SEISMIC CATEGORY I STRUCTURES, SYSTEMS AND COMPONENTS NOTES

1. These items not required as mechanical supports for CRDM housings, but are required to ensure functioning of the control rods.

2. Any reactor vessel internal, the single failure of which could cause release of a mechanical piece having potential for direct damage (as to the vessel cladding) or flow blockage, shall be classified to a minimum of Safety Class 2 (see Subsection 3.2.2.1 for definition), seismic Category I.

3. Failure could cause a loss-of-coolant accident, but less than a Condition III loss of coolant.

4. All seismic Category I structures are founded either on sound bedrock or on engineered backfill extending to sound bedrock. The type of engineered backfill used beneath the foundations of all seismic Category I structures was fill concrete, except for safety-related electrical duct banks, electrical manholes and service water pipes which were founded on offsite borrow or tunnel cuttings, as shown in Table 2.5-20.

5. The conduit and cable tray raceway systems including their supports, when used to carry safety-related circuit cables and other raceway installations whose failure during a seismic event could damage other safety-related systems or components are seismically qualified as assemblies. The items that make up the supports and cable trays are treated as nonsafety-related structural members, but are purchased as a component with specific performance requirements. The manufacturer provides substantiating test data and calculations, as well as a certificate of compliance to his manufacturing standards. The supports are assembled, installed and inspected in accordance with the applicable criteria of 10 CFR 50 Appendix B. The cable trays are designed, installed and inspected in accordance with Regulatory Position C.2 and C.4 of Regulatory Guide 1.29, Revision 3.

   Qualification of the conduit and cable tray raceways for the Class 1E safety related circuits have been confirmed by analysis and/or testing, both of these methods verify the adequacy of the system based on the properties of the raceways (including tray where applicable) and support components.

6. Components that are part of the diesel package but not essential to the operation i.e., electric motors for auxiliary coolant pump and auxiliary lube oil pump etc., are not included in this category.

7. See UFSAR Subsection 8.3.1.1.a4 for exception pertaining to the protection of the 13.8 kV containment electrical penetrations.

TABLE 3.2-2 SEISMIC AND SAFETY CLASSIFICATIONS FLUID SYSTEMS AND COMPONENTS

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
	Reactor Coolant System							
5.3	**Reactor Vessel**							
	Vessel and Closure Head	1	ASME III(7)	1	I	CS	W	
	Full Length CRDM Housing	1	ASME III	1	I	CS	W	
	CRDM Head Adapter Plugs	1	ASME III	1	I	CS	W	
	Incore Instrumentation Guide Tubes	1	ASME III	1	I	CS	W	
5.4.1	**Reactor Coolant Pump**							
	Pump Casing	1	ASME III	1	I	CS	W	
	Thermal Barrier	1	ASME III	1	I	CS	W	
	Thermal Barrier Heat Exchanger	1	ASME III	1	I	CS	W	
	No. 1 Seal Housing	1	ASME III	1	I	CS	W	
	No. 2 Seal Housing	2	ASME III	1	I	CS	W	See Note 4.
	Pressure Retaining							
	Bolting	1	ASME III	1	I	CS	W	
	Pump Flywheel	3	N/A	N/A	I	CS	W	
5.4.2	**Steam Generator**							
	Tube (Reactor Coolant) Side	1	ASME III	1	I	CS	W	
	Shell (Steam) Side	2	ASME III	1	I	CS	W	See Note 4.

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
5.4.3	**Reactor Coolant Piping**							
	Hot and Cold Leg Piping, Fittings and Fabrication	1	ASME III	1	I	CS	W	For safety class for other piping and associated valves in the Reactor Coolant System and other auxiliary systems, see Note 3.
	Crossover Leg Piping, Fittings and Fabrication	1	ASME III	1	I	CS	W	
	Surge Pipe, Fittings and Fabrication	1	ASME III	1	I	CS	W	
	Reactor Coolant Thermowells	1	ASME III	1	I	CS	W	
	Reactor Coolant System Vents	2	ASME III	1	I	CS	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
5.4.7	**Residual Heat Removal System**							
	Residual Heat Removal Pumps	2	ASME III	2	I	PB	W	See Notes 1a, 1c, and 2.
	Residual Heat Exchanger							
	Tube Side	2	ASME III	2	I	PB	W	
	Shell Side	3	ASME III	3	I	PB	W	See Note 1c.
	Piping and Valves							
	Reactor Coolant Pressure Boundary	1	ASME III	1	I	CS	AE	See Note 3.
	Sample/Drain Lines	NNS	ANSI B31.1	-	-	PB	AE	
	Other	2	ASME III	2	I	PB/CS	AE	See Note 3.
5.4.10	**Pressurizer**	1	ASME III	1	I	CS	W	See Note 1b.
5.4.11	**Pressurizer Relief Discharge System**							
	Pressurizer Relief Tank	NNS	ASME VIII	-	I	CS	W	
	Piping (Downstream of Safety Valves)	NNS	ANSI B31.1	-	I	CS	AE	
5.4.12	**Valves of Safety Class 1 to Safety Class 2 Interface**	1	ASME III	1	I	CS	W	
	Valves of Safety Class 2	2	ASME III	2	I	PB/CS	W	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
5.4.13	**Safety and Relief Valves**							
	Pressurizer Safety Valves	1	ASME III	1	I	CS	W	
	Pressurizer Power-Operated Relief Valves	1	ASME III	1	I	CS	W	
	Block Valves for Power-Operated Relief Valves	1	ASME III	1	I	CS	W	
	Piping Upstream of Pressurizer Safety and Relief Valves	1	ASME III	1	I	CS	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
	Engineered Safety Features							
6.2.2	**Containment Heat Removal Systems**							
	Containment Spray System							
	Containment Spray Pumps	2	ASME III	2	I	PB	AE	
	Spray Additive Tank	3	ASME III	3	I	WB	AE	
	Containment Spray Heat Exchangers							
	Tube Side	2	ASME III	2	I	PB	AE	
	Shell Side	3	ASME III	3	I	PB	AE	
	Refueling Water Storage Tank	2	ASME III	2	I	WB	AE	
	Spray Headers and Nozzles	2	ASME III	2	I	CS	AE	
	Piping and Valves	2	ASME III	2	I	CS/PB/WP	AE	See Note 3.
		3	ASME III	3	I	CS/PB	AE	
6.2.4	**Containment Isolation System**							
	Penetration Piping and Isolation Valves included as part of the Containment Isolation System	2	ASME III	MC	I	CS/CE	AE	(22)

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building[11]	Supplier	Notes[14]
6.2.5	**Combustible Gas Control in Containment**							
	Piping and Valves(pressurized)	2	ASME III	2	I	CS	AE	
	Containment Recirculating Filter System							
	Fans	3	AMCA	-	I	CS	AE	
	Filter Unit	NNS	ANSI N509/ N510	-	-	CS	AE	
	Ductwork and Dampers	3	SMACNA, ANSI N509/N510	-	I	CS	AE	
	Containment Purge System							
	All pressurized components upstream of emergency exhaust filters (flow meter, throttle valve, piping)	2	ASME III	2	I	CS/CE	AE	
	Containment Atmosphere Sample System							
	Supply and Exhaust Piping	2	ASME III	2	I	CS/CE	AE	
	Supply and Exhaust Valves	2	ASME III	2	I	CS/CE	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
6.3	**Emergency Core Cooling System**							
	Accumulator	2	ASME III	2	I	PB	W	
	Safety Injection Pump	2	ASME III	2	I	PB	W	See Notes 1a, 1c & 2.
	Piping							
	Reactor Coolant Pressure Boundary	1	ASME III	1	I	CS/PB	W	See Note 3.
	Drain, vent and test lines beyond the first pressure boundary isolation valve.	NNS	ANSI B31.1	-	-	PB	AE	
	Valves	2	ASME III and ANSI B16.5	2	I	PB/CS	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
6.5.1	**Engineered Safety Feature Filter Systems**							
	Containment Enclosure Emergency Cleanup System							
	Fans	2	AMCA	-	I	CE	AE	
	Filters (HEPA & Charcoal)	2	ANSI N509/ N510	-	I	CE	AE	
	Ductwork and Dampers	2	ANSI N509/ N510	-	I	CE	AE	
	Fuel Storage Building Emergency Cleanup System							
	Fans	3	AMCA	-	I	FB	AE	
	Filters (Charcoal & HEPA)	3	ANSI N509/ N510	-	I	FB	AE	
	Ductwork & Dampers	3	ANSI N509/ N510	-	I	FB	AE	
	Control Room Emergency Filtration Subsystem							
	Fans	3	AMCA	-	I	CD	AE/PSNH	
	Filters (HEPA & Charcoal)	3	ANSI N509/ N510	-	I	CD	AE/PSNH	
	Ductwork and Dampers	3	ANSI N509/ N510	-	I	CD	AE/PSNH	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
6.8	**Emergency Feedwater System**							
	Emergency Feedwater Pumps	3	ASME III	3	I	EF	AE	
	Piping and Valves	3	ASME III	3	I	EF	AE	
	Fans	3	AMCA	-	I	EF	AE	
	Ductwork	3	SMACNA	-	I	EF	AE	
	Dampers	3	AMCA					
	<u>Auxiliary Systems</u>							

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building[11]	Supplier	Notes[14]
9.1.3	**Spent Fuel Pool Cooling and Cleanup System**							
	Spent Fuel Pool Cooling Pump	3	ASME III	3	I	FB	AE	
	Spent Fuel Pool Skimmer Pump	NNS	MFRS. STDS.	-	-	FB	AE	
	Spent Fuel Pool Heat Exchanger	3	ASME III	3	I	FB	AE	
	Spent Fuel Pool Demineralizer	NNS	ASME VIII	-	-	PB	AE	
	Spent Fuel Pool Pre-Filter	NNS	MFRS. STDS.	-	-	PB	AE	
	Spent Fuel Pool Post- Filter	NNS	MFRS. STDS.	-	-	PB	AE	
	Spent Fuel Pool Skimmer	NNS	MFRS. STDS.	-	-	FB	AE	
	Spent Fuel Pool Strainer	NNS	MFRS. STDS.	-	-	FB	AE	
	Refueling Canal Skimmer Pump	NNS	MFRS. STDS.	-	-	CS	AE	
	Piping and Valves							
	Spent Fuel Pool Cooling Loop	3	ASME III	3	I	FB	AE	See Note 3.
	Spent Fuel Pool Skimmer Loop	NNS	ANSI B31.1	-	-	FB/PB	AE	
	Refueling Canal Cleanup Loop	NNS	ANSI B31.1	-	-	CS	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building[11]	Supplier	Notes[14]
9.2.1	**Station Service Water System**							
	Service Water Pumps	3	ASME III	3	I	CW	AE	
	Piping and Valves	3	ASME III	3	I	SW/PB/YD	AE	
	Strainers	3	ASME III	3	I	PB	AE	
9.2.2	**Primary Component Cooling Water System (PCCW)**							
	PCCW Pumps	3	ASME III	3	I	PB	AE	
	PCCW Heat Exchanger	3	ASME III	3	I	PB	AE	
	PCCW Head Tank	3	ASME III	3	I	PB	AE	
	Piping and Valves							
	Furnish and support cooling water supply to safeguards components	2/3	ASME III	2/3	I	CS/CF/FS	AE	See Note 3.
	Containment Penetration	2	ASME III	2	I	CE	AE	
	Other	NNS	ANSI B31.1	-	-	PB/CS WB/FB/YD	AE	
	Thermal Barrier							
	Pumps	3	ASME III	3	I	CS	AE	
	HX	2/3	ASME III	2/3	I	CS	AE	
	Head Tank	3	ASME III	3	I	CS	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.2.3	**Demineralized Makeup Water System**							
	Storage & Distribution Subsystem							
	Demineralized Water Storage Tanks	NNS	API-650, ASME VIII		-	YD	AE	
	Demineralized Water Transfer Pumps	NNS	MFRS. STDS.	-	-	TB	AE	
	Demineralized Water Storage Tank Recirc. Pump	NNS	MFRS. STDS.	-	-	CW	NHY	
	Demineralized Water Storage Tank Heat Exchangers	NNS	ASME VIII	-	-	TB/CW	AE/NHY	
	Piping and Valves	NNS	ANSI B31.1	-	-	PB/CD/TB	AE	
	Containment Penetration	2	ASME III	2	I	CS	AE	
	Water Treatment Subsystem	NNS	ASME VIII, MFRS. STDS.	-	-	TB	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.2.5	**Ultimate Heat Sink Atlantic Ocean**							
	Intake & Discharge Tunnels	NNS	-	-	-	-	AE	
	Piping	3	ASME III and	3	I	YD	AE	
		NNS	ANSI B31.1	-	-	YD/TB	AE	
	Expansion Joints	3	ASME III and	-	-	YD	AE	
		NNS	ANSI B31.1	3	1	PA/SW	AE	
	Cooling Tower							
	Cooling Tower Fans	3	MFRS. STDS.	-	I	CT	AE	
	Cooling Tower Pumps	3	ASME III	3	I	CT	AE	
	Piping and Valves	3	ASME III	3	I	CT	AE	
9.2.6	**Condensate Storage Facilities**							
	Condensate Storage Tank	3	ASME III	3	I	YD	AE	
	Piping and Valves	NNS	ANSI B31.1	-	-	YD/TB	AE	
9.2.7	**Reactor Makeup Water System**							
	Reactor Makeup Water	NNS	API-650,	-	-	YD	AE	
	Storage Tank		ASME VIII					
	Reactor Makeup Water Pump	NNS	ANSI B73.1	-	-	PB	AE	
	Piping and Valves							
	Containment Penetration	2	ASME III	2	I	CS	AE	
	Other	NNS	ANSI B31.1	-	-	PB/WB/CS	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.3.1	**Compressed Air System**							
	Air Compressor	NNS	ASME VIII	-	-	TB	AE	
	After-Cooler	NNS	ASME VIII	-	-	TB	AE	
	Moisture Separator	NNS	ASME VIII	-	-	TB	AE	
	Instrument Air Dryer Unit	NNS	ASME VIII	-	-	TB	AE	
	Air Receiver	NNS	ASME VIII	-	-	TB	AE	
	Filters	NNS	ASME VIII	-	-	TB	AE	
	Piping and Valves							
	Containment Penetration	2	ASME III	2	I	CE	AE	
	Other	NNS	ANSI B31.1	-	-	A11	AE	
	Vent Line from Safety-Related Devices to Class 1E Solenoid Vent Valve							
	Tubing	NNS	ANSI B31.1	-	I	All	AE	
	Flexible Connectors	NNS	MFR STD	-	I	All	AE	
	Safety-Related Backup Gas Supply							
	Tubing and Valves	3	ANSI B31.1	-	I	All	AE	
	Flexible Connectors	3	ANSI B31.1	-	I	All	AE	
	Gas Cylinders	3	DOT 3AA	-	I	All	AE	
	Pressure Regulator	3	MFR STD	-	I	All	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.3.2	**Process Sampling System**							
	Sample Heat Exchangers	NNS	ASME VIII	-	-	PB	AE	
	Sample Pressure Vessels	NNS	ASME VIII	-	-	PB	AE	
	Piping, Tubing and Valves Originating within Containment up to Isolation Valve	2	ASME III	2	I	CS/CE	AE	See Note 21.
	Post-Accident Sample Panel	NNS	No Code	-	-	PB	AE	
	Post-Accident Sample Pumps	NNS	MFRS. STDS.	-	-	PB	AE	
	Other	NNS	ANSI B31.1	-	-	PB	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.3.3	**Equipment and Floor Drainage System**							
	Reactor Coolant Drain Tank (RCDT)	NNS	ASME VIII	-	-	CS	AE	
	RCDT Pump	NNS	No Code	-	-	CS	AE	
	RCDT Heat Exchanger	NNS	ASME VIII	-	-	CS	AE	
	Sump Pumps	NNS	No Code	-	-	A11	AE	
	Chemical Drain Tank	NNS	ASME VIII	-	-	AB	AE	
	Chemical Drain Treatment Tank	NNS	ASME VIII	-	-	WB	AE	
	Chemical Drain Transfer Pump	NNS	No Code	-	-	AB	AE	
	Chemical Drain Treatment Pump	NNS	No Code	-	-	WB	AE	
	Piping and Valves							
	Containment Penetration	2	ASME III	2	I	CS	AE	
	Other	NNS	ANSI B31.1	-	-	A11	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building[11]	Supplier	Notes[14]
9.3.4	**Chemical and Volume Control System**							
	Centrifugal Charging Pump	2	ASME III	2	I	PB	W	See Notes 1a and 1c or 1d and 2.
	Positive Displacement Pump	2	ASME III	2	I	PB	W	See Notes 1a and 1c or 1d and 2.
	Boric Acid Transfer Pump	3	ASME III	3	I	PB	W	See Note 1b.
	Volume Control Tank	2	ASME III	2	I	PB	W	
	Boric Acid Tank	3	ASME III	3	I	PB	AE	
	Boric Acid Batching Tank	NNS	ASME VIII	-	-	PB	AE	
	Chemical Mixing Tank	NNS	ASME VIII	-	-	PB	AE	
	Resin Fill Tank	NNS	ASME VIII	-	-	PB	AE	
	Regenerative Heat Exchanger	2	ASME III	2	I	CS	W	
	Letdown Heat Exchanger					PB	W	
	Tube Side	2	ASME III	2	I			
	Shell Side	3	ASME III	3	I			See Note 2.
	Excess Letdown Heat Exchanger					CS	W	
	Tube Side	2	ASME III	2	I			
	Shell Side	3	ASME III	3	I			
	Seal Water Heat Exchanger					PB	W	
	Tube Side	2	ASME III	2	I			
	Shell Side	3	ASME III	3	I			See Note 2.
	Mixed Bed Demineralizer	3	ASME III	3	I	PB	AE	
	Cation Bed Demineralizer	3	ASME III	3	I	PB	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.3.4 (cont)	Boric Acid Blender	3	ASME III	3	I	PB	AE	
	Batching Tank Agitator	NNS	MFRS. STDS.	-	-	PB	AE	
	Letdown Degasifier	NNS	ASME VIII	-	-	PB	AE	
	Letdown Degasifier	NNS	ASME VIII	-	-	PB	AE	
	Preheater (Tube Side)							
	Letdown Degasifier	NNS	ASME VIII	-	-	PB	AE	
	Regenerative Heater							
	Letdown Valves	2	ASME III	2	I	CS	AE	
	Reactor Coolant Pump Seal	1	ASME III	1	I	CS	AE	
	Bypass Orifice							
	Boric Acid Transfer Pump	3	ASME III	3	I	PB	AE	
	Bypass Orifice							
	Chemical Mixing Tank Orifice	NNS	ASME VIII	-	-	PB	AE	
	Reactor Coolant Filter	2	ASME III	2	I	PB	AE	
	Seal Water Injection Filter	2	ASME III	2	I	PB	AE	
	Seal Water Return Filter	2	ASME III	2	I	PB	AE	
	Boric Acid Filter	3	ASME III	3	I	PB	AE	
	Demineralized Prefilter	3	ASME III	3	I	PB	AE	
	Boron Meter	NNS	ANSI B31.1	-	-	PB	W	Classified on basis that flow restriction is provided in the piping.
	Reactor Coolant Pump Standpipe	NNS	ASME VIII	-	-	CS	W	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
	Reactor Coolant Pump Standpipe Orifice	NNS	-	-	-	CS	W	
9.3.4 (cont)	**Chemical and Volume Control System**							
	Piping	2	ASME III	2	I	CS/PB	W/AE	See Note 3.
		3	ASME III	3	I	PB	W/AE	
		NNS	ANSI B31.1	-	-	PB	AE	
	Valves	2	ASME III	2	I	CS/PB	W/AE	See Note 3.
		3	ASME III	3	I	CS/PB	W/AE	
		NNS	ANSI B31.1	-	-	CS/PB	AE	
	Boron Thermal Regeneration Subsystem							
	Chiller Pump	NNS	No Code	-	-	PB	W	See Note 6.
	Chiller Surge Tank	NNS	ASME VIII	-	-	PB	W	See Note 6.
	Moderating Heat Exchanger					PB	W	
	Tube Side	3	ASME III	3	-			See Note 6.
	Shell Side	3	ASME III	3	-			See Note 6.
	Letdown Chiller Heat Exchanger					PB	W	
	Tube Side	3	ASME III	3	-			See Note 6.
	Shell Side	NNS	ASME VIII	-	-			See Note 6.
	Letdown Reheat Heat Exchanger					PB	W	
	Tube Side	2	ASME III	2	I			See Note 6.

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building[11]	Supplier	Notes[14]
	Shell Side	3	ASME III	3	-			See Note 6.
9.3.4 (cont)	Thermal Regeneration	3	ASME III	3	-	PB	W	See Note 6.
	Demineralizer							
	Chiller Unit							W
	Evaporator	NNS	ASME VIII	-	-			See Note 6.
	Condenser	NNS	ASME VIII	-	-			See Note 6.
	Compressor	NNS	No Code	-	-			See Note 6.

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.3.5	**Boron Recovery System**							
	Recovery Evaporator Feed Pump	NNS	No Code	-	-	WB	AE	
	Recovery Test Tanks	NNS	API-650, ASME VIII	-	-	WB	AE	See Note 8.
	Recovery Test Tank Pump	NNS	No Code	-	-	WB	AE	
	Boron Waste Storage Tanks	NNS	API-620, ASME VIII	-	-	WB	AE	See Note 8.
	Recovery Evaporator Package							
	Recovery Evaporator	NNS	ASME VIII	-	-	WB	AE	See Note 8.
	Bottoms Pump	NNS	No Code	-	-	WB	AE	See Note 8.
	Bottoms Cooler	NNS	ASME VIII	-	-	WB	AE	See Note 8.
	Reboiler	NNS	ASME VIII, TEMA C	-	-	WB	AE	See Note 8.
	Reboiler Pump	NNS	No Code	-	-	WB	AE	See Note 8.
	Distillate Pump	NNS	No Code	-	-	WB	AE	See Note 8.
	Distillate Accumulator	NNS	ASME VIII	-	-	WB	AE	See Note 8.
	Distillate Condenser	NNS	ASME VIII	-	-	WB	AE	See Note 8.
	Distillate Cooler	NNS	ASME VIII, TEMA C	-	-	WB	AE	See Note 8.
	Cesium Removal Ion Exchanger	NNS	ASME VIII	-	-	WB	AE	See Note 8.
	Recovery Demineralizer	NNS	ASME VIII	-	-	WB	AE	See Note 8.
	Primary Drain Tank (PDT)	NNS	ASME VIII	-	-	WB	AE	
	PDT Transfer Pump	NNS	MFRS. STDS.	-	-	WB	AE	
	PDT Degasifier	NNS	ASME VIII	-	-	WB	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building[11]	Supplier	Notes[14]
9.3.5 (Cont)	PDT Degasifier Recirc. Pump	NNS	MFRS. STDS.	-	-	WB	AE	
	PDT Degasifier Preheater	NNS	ASME VIII, TEMA R	-	-	WB	AE	
	PDT Degasifier Regenerative Heat Exchanger	NNS	ASME VIII, TEMA R	-	-	WB	AE	
	PDT Degasifier Trim Cooler	NNS	ASME VIII, TEMA R	-	-	WB	AE	
	Filters	NNS	ASME VIII	-	-	WB	AE	
	Piping and Valves	NNS	ANSI B31.1	-	-	WB	AE	
9.3.6	**Equipment Vent System**							
	Safety Valve Surge Tank	NNS	ASME VIII	-	-	PB	AE	
	Evacuation Pump	NNS	MFRS. STDS.	-	-	CS	PSNH	
	Separator/Silencer	NNS	ASME VIII	-	-	CS	PSNH	
	Piping and Valves							
	Containment Penetration	2	ASME III	2	I	CS	AE	
	Other	NNS	ANSI B31.1	-	-	WB/PB/CS	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.4.1	**Control Room Complex Ventilation System**							
	Control Room Air Conditioning Subsystem							
	Air Conditioning/Fan	3	MFRS. STDS.	-	I	CD	AE	
	Water Chiller	3	MFRS. STDS.	-	I	CD	AE	
	Dampers	3	AMCA	-	I	CD	AE	
	Ductwork	3	SMACNA	-	I	CD	AE	See Note 13.
	Computer Room Air Conditioning Subsystem							
	Air Conditioning Unit	-	MFRS. STDS.	-	-	CD	AE	
	Condensing Unit	-	MFRS. STDS.	-	-	CD	AE	
	Cooling Coil	3	ARI 410	-	I	CD		
	Chilled Water Pumps	3	MFRS. STDS.	-	I	CD	AE	
	Expansion Tank	3	ASME VIII	-	I	CD	AE	
	Piping and Valves	3	ANSI B31.1	-	I	CD	AE	
	Dampers	-	AMCA	-	-	CD	AE	
	Ductwork	-	SMACNA	-	-	CD	AE	See Note 20.
	Control Room Complex Normal Makeup Air Subsystem							
	Fans	3	AMCA	-	I	CD	AE	
	Dampers	3	AMCA	-	I	CD	AE	
	Intake Piping	3	ANSI B31.1	NNS__ 1A	I	CD/YD	AE	See Note 19.

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.4.1 (Cont.)	Control Room Emergency Filtration Subsystem							
9.4.2	**Fuel Storage Building Ventilation System**							
	Ventilation Fans	-	AMCA	-	-	FB	AE	
	Dampers	3	AMCA	-	I	FB	AE	
	Ductwork	-	SMACNA	-	I	FB	AE	
9.4.3	**Primary Auxiliary Building Ventilation System**							
	Ventilation Fans		AMCA	-	-	PB	AE	
	PCCW and BI Pump Area Wall		AMCA	-	I	PB	AE	
	Fans and Dampers Normal Cleanup Exhaust Fans		AMCA	-	-	PB	AE	
	Normal Cleanup Filters		ANSI N509/ N510	-	-	PB	AE	
	Ductwork and Dampers Isolation Dampers and Ductwork Passing through CE	2	ANSI N509/N510	-	I	PB/CE	AE	
	Other	NNS	SMACNA	-	I	PB	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building[11]	Supplier	Notes[14]
9.4.4	**Waste Processing Building Ventilation System**							
	Fans	-	AMCA	-	-	WB	AE	
	Ductwork	-	SMACNA	-	-	WB	AE	
	Dampers	-	AMCA	-	-	WB	AE	
	Filters	-	ANSI N509/N510	-	-	WB	AE	
9.4.5	**Containment Ventilation System**							
	Recirculation Fan/Coolers	-	MFRS. STDS.	-	I	CS	AE	See Note 9.
	Ductwork	-	SMACNA	-	I	CS	AE	See Note 10.
9.4.6	**Containment Enclosure Ventilation System**							
	Fans	3	AMCA	-	I	CE	AE	
	Cooling Coils	3	ASME III	-	I	CE	AE	
	Ductwork	3	ANSI N509/ N510	-	I	CE	AE	
9.4.7	**Electrical Penetration Area Air Conditioning System**							
	Air Conditioning Unit	-	MFRS. STDS.	-	-	EF	AE	
	Condensing Unit	-	MFRS. STDS.	-	-	EF	AE	
	Ductwork	-	SMACNA	-	-	EF	AE	
	Dampers	-	AMCA	-	-	EF	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.4.8	**Diesel Generator Building Ventilation System**							
	Fans	3	AMCA	-	I	CD	AE	
	Ductwork	3 (Mod)	SMACNA	-	I	CD	AE	
	Dampers	3	AMCA	-	I	CD	AE	
9.4.9	**Cable Spreading Room Ventilation System**							
	Fans	-	AMCA	-	-	CD	AE	
	Ductwork	-	SMACNA	-	I	CD	AE	
	Dampers	-	AMCA	-	I	CD	AE	
9.4.10	**4 kV Switchgear Area and Battery Rooms Ventilation System**							
	Fans	3	AMCA	-	I	CD	AE	
	Dampers	3	AMCA	-	I	CD	AE	
	Ductwork	3 (Mod)	SMACNA	-	I	CD	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.4.11	**Emergency Feedwater Pumphouse Ventilation System**							
	Fans	3	AMCA	-	I	EF	AE	
	Dampers	3	AMCA	-	I	EF	AE	
	Ductwork	3 (Mod)	SMACNA	-	I	EF	AE	
9.4.13	**Service Water Pumphouse Ventilation System**							
	Fans	3	AMCA	-	I	CW	AE	
	Dampers	3	AMCA	-	I	CW	AE	
	Ductwork	3 (Mod)	SMACNA	-	I	CW	AE	
9.4.14	**Service Water Cooling Tower Pump Room and Switchgear Room Ventilation Systems**							
	Pump Room Power Roof Ventilator	3	AMCA	-	-	CT	AE	
	Switchgear Room Supply Fan	3	AMCA	-	I	CT	AE	
	Ductwork	3 (Mod)	SMACNA	-	I	CT	AE	
	Dampers	3	AMCA	-	I	CT	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.5.1	**Fire Protection System**							
	Water Supply Tank	-	NFPA STDS.	-	-	YD	AE	
	Fire Pump	-	NFPA STDS.	-	-	YD	AE	
	Jockey Pump	-	NFPA STDS.	-	-	YD	AE	
	Piping and Valves	-	NFPA STDS.	-	-	YD	AE	
9.5.4	**Diesel Generator Fuel Oil Storage and Transfer System**							
	Fuel Oil Day Tank	3	ASME III	3	I	CD	AE	
	Fuel Oil Storage Tank	3	ASME III	3	I	CD	AE	
	Fuel Oil Transfer Pump	3	ASME III	3	I	CD	AE	
	Associated Piping and Valves for Above Tanks and Pump	3	ASME III	3	I	CD	AE	
	All Remaining On-Engine Equipment and Piping	3	MFRS. STDS.	-	I	CD	AE	See system P&ID for safety class boundaries.

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.5.5	**Diesel Generator Cooling Water System**							
	Auxiliary Coolant Pump	3	MFRS. STDS.	-	I	CD	AE	
	DG Component Cooling Water Heat Exchanger	3	ASME III	3	I	PB	AE	
	Expansion Tank	3	ASME III	3	I	CD	AE	
	Interconnecting Piping and Valves for Above Components, and On-Skid, Off-Engine Piping	3	ASME III	3	I	CD/PB/YD	AE	
	All Remaining Essential On-Engine Equipment and Piping	3	MFRS. STDS.	-	I	CD	AE	See system P&ID for safety class boundaries.
9.5.6	**Diesel Generator Starting System**							
	Air Receivers	3	ASME III	3	I	CD	AE	
	Air Compressor	3	MFRS. STDS.	-	I	CD	AE	
	Piping and Valves from Air Receiver to DG Skid	3	ASME III	3	I	CD	AE	
	All Remaining Essential Equipment Piping	3	MFRS. STDS.	-	I	CD	AE	See system P&ID for safety class boundaries.
	Backup Control Air Compressor	3	MFRS. STDS.	-	I	CD		

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
9.5.7	**Diesel Generator Lubrication System**							
	Auxiliary Lube Oil Pump	3	MFRS. STDS.	-	I	CD	AE	
	Associated Piping and Valves for Above Pump	3	ASME III	3	I	CD	AE	
	All Remaining Essential On-Engine Equipment and Piping	3	MFRS. STDS.	-	I	CD	AE	See system P&ID for safety class boundaries.
9.5.8	**Diesel Generator Combustion Air Intake and Exhaust System**							
	Piping (Off-Engine)	3	ANSI B31.1	-	I	CD	AE	
	Air Intake Filter	3	MFRS. STDS.	-	I	CD	AE	
	Exhaust Silencer	3	MFRS. STDS.	-	I	CD	AE	
	Exhaust Expansion Joints	3	MFRS. STDS.	-	I	CD	AE	
	Piping (On-Engine)	3	MFRS. STDS.	-	I	CD	AE	
	Steam & Power Conversion System							

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
10.3	**Main Steam System**							
	Main Steam Piping and Valves from Steam Generator up through Containment Isolation Valves	2	ASME III	2	I	MF/CS	AE	
	To Emergency Feedwater Pump Turbine Interface	3	ASME III	3	I	MF/EF	AE	
	EFW Pump Turbine Assembly	3	MFRS. STDS.	-	I	MF/EF	AE	
	Other	NNS	ANSI B31.1	-	-	TB	AE	
10.4.2	**Main Condenser Evacuation**							
	System							
	Vacuum Pump	NNS	MFRS. STDS.	-	-	TB	AE	
	Piping and Valves	NNS	ANSI B31.1	-	-	TB	AE	
10.4.3	**Turbine Gland Sealing System**							
	Gland Steam Condenser and Exhauster Fans	NNS	MFRS. STDS.	-	-	TB	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
10.4.7	**Condensate and Feedwater System**							
	Condensate Pump	NNS	MFRS. STDS.	-	-	TB	AE	
	Heater Drain Pump	NNS	MFRS. STDS.	-	-	TB	AE	
	Steam Generator Feed Pump	NNS	MFRS. STDS.	-	-	TB	AE	
	Steam Generator Startup Feed Pump	NNS	MFRS. STDS.	-	-	TB	AE	
	Feedwater Heater	NNS	ASME VIII	-	-	TB	AE	
	Piping and Valves							
	From First Isolation Valve Outside Containment to Steam Generator	2	ASME III	2	I	MF/CS	AE	
	Other	NNS	ANSI B31.1	-	-	TB	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building[11]	Supplier	Notes[14]
10.4.8	**Steam Generator Blowdown System**							
	Flash Tank	NNS	ASME VIII	-	-	PB	AE	
	Flash Tank Bottoms Cooler	NNS	ASME VIII, TEMA C	-	-	PB	AE	
	Flash Steam Condenser/ Cooler	NNS	ASME VIII, TEMA C	-	-	PB	AE	
	Flash Tank Distillate Pump	NNS	MFRS. STDS.	-	-	PB	AE	
	Blowdown Evaporator					PB	AE	See Note 8.
	Vapor Body	NNS	ASME VIII	-	-			
	Heating Element	NNS	ASME VIII, TEMA C	-	-			
	Distillate Condenser	NNS	ASME VIII, TEMA C	-	-			
	Distillate Accumulator	NNS	ASME VIII, TEMA C	-	-			
	Distillate Pump	NNS	MFRS. STDS.	-	-			
	Distillate Cooler	NNS	ASME VIII, TEMA C					
	Bottoms Pump	NNS	MFRS. STDS.	-	-			
	Bottoms Cooler	NNS	ASME VIII	-	-			
	Piping and Valves Inside Containment, up through Isolation Valves	2	ASME III	2	I	CS	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
10.4.8 (Cont.)	Other Piping and Valves	NNS	ANSI B31.1	-	-	PB	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
	Radioactive Waste Management							
11.2	**Liquid Waste System**							
	Waste Evaporator Package							
	Reboiler	NNS	ASME VIII,	-	-	WB	AE	See Note 8.
			TEMA C					
	Reboiler Pump	NNS	MFRS. STDS.	-	-	WB	AE	See Note 8.
	Bottoms Pump	NNS	MFRS. STDS.	-	-	WB	AE	See Note 8.
	Bottoms Cooler	NNS	ASME VIII	-	-	WB	AE	See Note 8.
	Waste Evaporator	NNS	ASME VIII	-	-	WB	AE	See Note 8.
	Distillate Pump	NNS	MFRS. STDS.	-	-	WB	AE	See Note 8.
	Distillate Accumulator	NNS	ASME VIII	-	-	WB	AE	See Note 8.
	Distillate Cooler	NNS	ASME VIII, TEMA C	-	-	WB	AE	See Note 8.
	Distillate Condenser	NNS	ASME VIII, TEMA C	-	-	WB	AE	See Note 8.
	Floor Drain Tank	NNS	ASME VIII	-	-	WB	AE	See Note 8.
	Waste Test Tank	NNS	ASME VIII	-	-	WB	AE	See Note 8.
	Duplex Strainer	NNS	MFRS. STDS.	-	-	WB	AE	
	Waste Test Tank Pump	NNS	MFRS. STDS.	-	-	WB	AE	
	Floor Drain Tank Pump	NNS	MFRS. STDS.	-	-	WB	AE	
	Waste Demineralizer	NNS	ASME VIII	-	-	WB	AE	See Note 8.
	Waste Demineralizer Filter	NNS	ASME VIII	-	-	WB	AE	
	Floor Drain Tank Filter	NNS	ASME VIII	-	-	WB	AE	
	Piping and Valves	NNS	ANSI B31.1	-	-	WB	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
11.3	**Gaseous Waste System**							
	Carbon Delay Beds	NNS	ASME III	-	-	WB	AE	
	Iodine Guard Beds	NNS	ASME III	-	-	WB	AE	
	Gas Chiller	NNS	ASME III	-	-	WB	AE	
	Waste Gas Dryer	NNS	ASME III	-	-	WB	AE	
	Hydrogen Surge Tank	NNS	ASME VIII	-	-	WB	AE	
	Purge Gas Condenser	NNS	ASME VIII	-	-	WB	AE	
	Gas Chiller Compressor Unit	NNS	MFRS. STDS.	-	-	WB	AE	
	Hydrogen Gas Compressor Package	NNS	MFRS. STDS.	-	-	WB	AE	
	Regenerative Compressor Unit Package	NNS	MFRS. STDS.	-	-	WB	AE	
	Piping and Valves in process portion of piping up to and including the carbon delay beds	NNS	ASME III	-	-	WB	AE	See Note 14.
	Carbon delay bed support element up to and including the attachment weld	NNS	AISC	-	I	WB	AE	See Note 17.
	Other Piping and Valves	NNS	ANSI B31.1	-	-	WB	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
11.4	**Solid Waste System**							
	Spent Resin Sluice Tank	NNS	ASME VIII	-	-	WB	AE	
	Spent Resin Sluice Pump	NNS	MFRS. STDS.	-	-	WB	AE	
	Spent Resin Sluice Filter	NNS	ASME VIII	-	-	WB	AE	
	Spent Resin Transfer Pump	NNS	MFRS. STDS.	-	-	WB	AE	
	Waste Concentrates Tank	NNS	ASME VIII	-	-	WB	AE	
	Waste Concentrates Transfer Pump	NNS	MFRS. STDS.	-	-	WB	AE	
	Resin Hopper	NNS	ASME VIII	-	-	WB	AE	
	Spent Resin Dewatering Pump	NNS	MFRS. STDS.	-	-	WB	AE	
	Resin Centrifuge Metering Pump	NNS	MFRS. STDS.	-	-	WB	AE	
	Resin Centrifuge	NNS	MFRS. STDS.	-	-	WB	AE	
	Crystallizer Skid							
	Vessels	NNS	ASME VIII	-	-	WB	AE	
	Heat Exchangers	NNS	TEMA R	-	-	WB	AE	
	Pump	NNS	MFRS. STDS.	-	-	WB	AE	
	Extruder/Evaporator	NNS	DIN	-	-	WB	AE	
	Asphalt Storage Tank	NNS	API 650	-	-	WB	AE	
	Auxiliary Boiler Vessel	NNS	ASME VIII	-	-	WB	AE	
	Asphalt Recirculation Pump	NNS	MFRS. STDS.	-	-	WB	AE	
	Asphalt Metering Pump	NNS	MFRS. STDS.	-	-	WB	AE	
	Extruder Vent Hood	NNS		-	-	WB	AE	

UFSAR Section	**Systems and Components**	**ANS Safety Class**	**Principal Design/Const. Codes/Stds.**	**Code Class**	**Seismic Category**	**Building**[11]	**Supplier**	**Notes**[14]
	Auxiliary Boiler Chemical Feed Skid	NNS	MFRS. STDS.	-	-	WB	AE	

UFSAR Section	Systems and Components	ANS Safety Class	Principal Design/Const. Codes/Stds.	Code Class	Seismic Category	Building(11)	Supplier	Notes(14)
11.4 (Cont.)	Auxiliary Boiler Blowdown Tank	NNS	MFRS. STDS.	-	-	WB	AE	
	Steam Condensate Collection Skid	NNS	MFRS. STDS.	-	-	WB	AE	
	Auxiliary Boiler Feed Pumps	NNS	MFRS. STDS.	-	-	WB	AE	
	Auxiliary Boiler	NNS	MFRS. STDS.	-	-	WB	AE	
	Condensate Return Tank Trash Compactor	NNS	MFRS. STDS.	-	-	WB	AE	
	Trash Compactor Filter	NNS	MFRS. STDS.	-	-	WB	AE	
	Trash Compactor	NNS	MFRS. STDS.	-	-	WB	AE	
	Hydraulic Unit Piping and Valves	NNS	ANSI B31.1	-	-	WB	AE	

NOTES

1. Services required to support a safety or other necessary function:
   a. Emergency power - automatic loading.
   b. Emergency power - manual loading.
   c. Component cooling water.
   d. Service water.
2. Portions of equipment containing component cooling water are SC-3, Code Class 3.
3. Safety classes for piping and valves are as defined by the system piping and instrumentation diagrams. Code classes are those required by the safety class.
4. Represents code class upgrading. As permitted by paragraph NA-2134 of the ASME Code, Section III, this component is upgraded from the minimum required Code Class 2 to Code Class 1.
5. Parts are mechanically of safety class and must meet the structural integrity requirements of the specification and quality assurance requirements of 10 CFR 50, Appendix B.
6. This component is Safety Class 3 under the definition 2.2.3(1), (3), or (4) of ANSI N18.2-1975 and qualifies for no special seismic design by meeting the four following conditions. Portions of systems in which this component is located that perform the same safety function likewise qualify for no special seismic design.

   Conditions to be met for exemption are the following:
   a. Failure would not directly cause a Condition III or IV event (as defined in "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2-1973).
   b. There is no safety function to mitigate nor could failure prevent mitigation of the consequences of a Condition III or IV event.
   c. Failure during or following any Condition II event would result in consequences no more severe than allowed for a Condition III event.
   d. Routine post-seismic procedures would disclose loss of the safety function

7. ASME: American Society of Mechanical Engineers. III stands for Section III of the ASME Boiler and Pressure Vessel Code (B&PVC). VIII stands for Section VIII of the ASME B&PVC. Pressure vessels which are part of the reactor coolant pressure boundary meet the requirements of the 1971 Version, with application of all Addenda through to and including the Summer 1972 Addenda. Pumps, valves and piping which are part of the reactor coolant pressure boundary meet the requirements of the 1971 Version, with application of all Addenda through to and including the Winter 1972 Addenda. Later code versions may be used optionally.

8. Augmented to provide additional quality assurance. Provisions include completely welded systems except where maintenance or testing requires flanged connections, material certifications consistent with ASME Section III, ND-2121, and mandatory hydrostatic testing of all systems.

9. Fan/cooling units are designed and seismically analyzed to assure that they will not overturn or fail structurally during an SSE.

10. Ductwork supports are designed and seismically analyzed to withstand an SSE where support failure could damage safeguards equipment.

11. Building code:

    AB = Administration and Service Building

    CE = Containment Enclosure Building

    CD = Control and Diesel Generator Building

    CS = Containment Structure

    CT = Service Water Cooling Tower

    CW = Service and Circulating Water Pumphouse

    EF = Auxiliary Feedwater House & Electrical Penetration Area

    FB = Fuel Storage Building

    PB = Primary Auxiliary Building

    MF = Main Steam and Feedwater Pipe Chase

    TB = Turbine Building

    WB = Waste Processing Building

    YD = Yard

Arrangement drawings for the buildings in which the systems are located are presented in Section 1.2.

12. Ductwork from the downstream side of the air cleaning units to the fan intakes and discharge of the fans to the building boundaries is Safety Class 3, seismic Category I.

13. Ductwork located within the mechanical equipment room to the boundary of the control room is Safety Class 3, seismic Category I

14. Motors, valve operators and valve actuators which must operate (run, open or close) in order for the system to perform its safety function are classified as within the scope of the OQAP. Motors or operators which are associated with mechanical components which serve only as part of a pressure boundary are not within the scope of the OQAP.

15. Deleted

16. (Deleted in Amendment 60)

17. The tank support elements should satisfy the requirements of Position 5 of Regulatory Guide 1.143, Rev. 1.

18. System specifications provided under Updated FSAR Subsection 6.5.1.

19. Remote air intake piping meets ANS/HVAC Safety Class 3 criteria and NNS-1A piping criteria.

20. Control room air conditioning system provides cooling air to the computer room in the event computer room air conditioning system fails.

21. Components will have same safety class/codes as process piping if connected to Safety Class 3 or lower classifications.

22. The CAP penetrations (HVAC-1, HVAC-2) and associated blind flanges are qualified to, and classified as, ASME III, subsection NE, Class MC.

TABLE 3.2-3 QUALITY STANDARDS [(1)]

Components[(2)]	Safety Class 1	Safety Class 2	Safety Class 3	Non Nuclear Safety (NNS)
Pressure Vessels	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 2	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 3	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1
Piping	As above	As above	As above	ANSI B31.1.0 Power Piping
Pumps	As above	As above	As above	ASME Boiler and Pressure Vessel Code, Sec. VIII, Div. 1 and Mfrs. Standards
Valves	As above	As above	As above	ANSI B31.1.0
Atmospheric Storage Tanks	-	As above	As above	API-650
0-15 psig Storage Tanks	-	As above	As above	API-620
Piping used as HVAC Duct	-	-	ANSI B31.1.0	ANSI B31.1.0

(1) Pressure vessels which are part of the reactor coolant pressure boundary meet the requirements of the 1971 Version of the ASME Boiler and Pressure Vessel Code, with application of all addenda through to and including the Summer 1972 Addenda. Pumps, valves and piping which are parts of the reactor coolant pressure boundary meet the requirements of the 1971 Version, with application of all addenda through to and including the Winter 1972 Addenda. Later code versions may be used optionally.

(2) Components other than those described in this table are controlled by quality standards discussed in the component descriptions contained in pertinent sections of the UFSAR.

TABLE 3.2-4 SEISMIC AND SAFETY CLASSIFICATIONS HVAC SYSTEMS AND COMPONENTS

			Ductwork		Supports	
Function Classification		**Symbol**	**Criteria**	**Guideline**	**Criteria**	**Guideline**
Safety- Related	Class 2 Seis. Cat. I	2 (Note 1)	Nonrigid Allowed	Rigid	Nonrigid Allowed	Rigid
	Class 3 Seis. Cat. I	3* (Note 2)	Nonrigid Allowed	Rigid	Nonrigid Allowed	Rigid
Nonsafety-Related	Seismic Category I	1	Nonrigid Allowed	Rigid SMACNA (Note 3)	Nonrigid Allowed	Rigid
	Nonseismic Nonsafety	-	-	SMACNA (Note 3)	-	SMACNA

Notes

1. These are 10 gauge welded ductwork leak tested to ANSI N509/N510 requirement.
2. These are either 10 gauge welded ductwork leak tested to ANSI N509/N510 requirements or in some specific instances upgraded SMACNA high pressure ductwork leak tested to SMACNA requirements.
3. These are standard SMACNA high pressure ductwork leak tested to SMACNA requirements.

TABLE 3.3-1 WIND PRESSURE ON CONTAINMENT ENCLOSURE BUILDING

Height Above	**Wind Velocity**	**q[1]**	**p = 0.5q[2]**
Grade (feet)	(mph)	(psf)	(psf)
0 – 50	110	37	19
50-150	135	56	28
Above 150	165	84	42

(Reference - ASCE Paper 3269)

(1) q - .00256 $(1.1V)^2$

(2) The full value of p is distributed horizontally by means of the pressure coefficients (CP) on Figure 3.3-1

TABLE 3.3-2 WIND VELOCITY PRESSURES

Height of Grade	q_F	q_p	q_M
	PSF	PSF	PSF
30' or less	40	46	31
Over 30' and up to 50'	46	51	36
Over 50' and up to 100'	53	59	44
Over 100' and up to 150'	58	65	49
Over 150' and up to 200'	62	69	53
Over 200' and up to 250'	65	72	57
(Reference - ANSI-A58.1-1972)			

TABLE 3.3-3 TORNADO WIND PRESSURE COMPONENTS

	ASCE	ANSI
W_{W1} = External Tornado Wind Pressure		
For Rectangular Structures		
Wind Wall		0.8 $q_{avg.}$
Leeward Wall		-0.5 $q_{avg.}$
Side Walls		-0.7 q_{avg}
Roof		-0.7 $q_{avg.}$
For Cylindrical Structures:	Cpe(2) x $q_{avg.}$(1)	Cpe x $q_{avg.}$
W_{w2} = Internal Tornado Wind Pressure	±0.2(3) $q_{avg.}$	±0.3 $q_{avg.}$

(1) $q_{avg.} = C_s\, q_{max.} = C_s$ (332) PSF. See Fig. 3.3-2 for C_s

(2) C_{pe} = External Pressure Coefficient. See Fig. 3.3-1 for C_{pe}

(3) Coefficient ±0.2 is for uniformly distributed openings. All other cases are evaluated from Table 4.0 of Ref. 1

TABLE 3.3-4 **NON-CATEGORY I STRUCTURES DESIGNED AGAINST COLLAPSE ONTO ADJACENT CATEGORY I STRUCTURES DUE TO TORNADO WIND LOADING**

Non-Category I Structures	**Affected Category I Structure or Component**
Turbine Building [(1)]	Condensate Storage Tank (E.W.), Containment and Others (N.S.)
Nonessential Switch gear Building [(2)]	Control and Diesel Generator Building
Tank Farm Area (Steel Framing Portion which includes Steel Framing, Concrete Roofing and Metal Siding over Refueling Water Storage Tank) [(3)]	Primary Auxiliary Building, Waste Processing Building and Tank Farm Area Tunnels
Steam Generator Blowdown Recovery Building [(4)]	Primary Auxiliary Building and Waste Process Building
Circulating Water Pumphouse Steel Framing Portion [(5)]	Service Water Pumphouse
Waste Processing Building Steel Framing Portion and reinforced concrete portion, except the area between Columns 1 & 2 and A to D between elevations 53'-0" and 86'-0"	Primary Auxiliary Building, Tank Farm Area and Piping Tunnels

NOTES:

(1) The entire Turbine Building is designed against failure in the north-south direction. The south end is designed against failure in the east-west direction; an east-west failure in the north end will not affect any seismic Category I structures.

(2) The Nonessential Switchgear Building is designed mechanistically to fall away from the Control and Diesel Generator Building under the action of a collapsing Administration and Service Building. Thus, no significant load is applied to the Control and Diesel Generator Building by either the falling Administration and Service Building or the falling Nonessential Switchgear Building.

(3) The steel framing portion has been evaluated for tornado wind loads against collapse on surrounding safety-related systems, components and structures and it has been found that there are no adverse effects. The tornado effects of the steel framing portion upon the systems and components located within the tank farm structure are not a design consideration because the loss of function of these systems and components will not affect the capability of a safe reactor shutdown.

(4) The collapse of the steel framing upon surrounding Category I structures and components as stated above was evaluated to prove that these category structures and components are not damaged to the extent of not being able to perform their function. The tornado effects of the steel framing portion upon the systems and components located within the Steam Generator Blowdown Recovery Building are not a design consideration because the loss of function of these systems and components will not affect the capability of a safe reactor shutdown.

(5) The collapse of the Circulating Water Pumphouse on the Service Water Pumphouse was evaluated to prove that the collapse will not impair the Service Water Pumphouse or system.

TABLE 3.5-1 **STRUCTURES, SHIELDS AND BARRIERS DESIGNED TO RESIST INTERNAL AND EXTERNAL MISSILES**

	Internal Missiles	**External Missiles**
Containment shell		1
Containment structure primary shield wall	2	
Secondary shield wall	2	
Pressurizer shielding	2	
Reactor missile shield	2	
Operating floor	2	
Primary Auxiliary Building	3	1
Refueling water storage tank	See Section 6.2	
Condensate storage tank		1
Control and Diesel Generator Building	3	1
Shield between diesel generators	3	
Fuel Storage Building	3	1
Electrical penetration tunnel		1
Service Water Pumphouse, intake and discharge transition structures, flume and backwash ducts	3	1
Emergency Feedwater Pump Building	3	1
Diesel fuel tanks		
Piping tunnels	3	1
Main steam and feedwater lines outside containment to 2nd isolation valve		1
Category I electrical manholes		1

1 - Tornado generated missiles

2 - Missiles postulated within the containment

3 - Temperature sensing element missile

TABLE 3.5-2 SUMMARY OF CONTROL ROD DRIVE MECHANISM MISSILE ANALYSIS

	Weight	**Thrust Area**	**Impact Area**	**Impact Velocity**	**Kinetic Energy**
Postulated Missiles	**(lb)**	**(in.^{2})**	**(in.^{2})**	**(ft/sec)**	**(ft-lb)**
Mechanism Housing Plug	6.5[a]	5.94	7.07	255	6563
Drive Shaft	170[b]	2.40	2.41	97.6	25147

a. Reflects Removal of Eyebolt

b. Heavy Drive Rod.

TABLE 3.5-3 VALVE - MISSILE CHARACTERISTICS

Missile Description	Weight (lb.)	Flow Discharge Area (in^2)	Thrust Area (in^2)	Impact Area (in^2)	Weight to Impact Area Ratio (psi)	Velocity (fps)
Safety Valve Bonnet	350	2.86	80	24	14.6	110
3 Inch Motor-Operated Isolation Valve Bonnet (plus motor operator and stem)	400	5.5	113	28	14.1	135
3 Inch Air-Operated Valve Bonnet (plus stem)	75	1.8	20	20	3.75	115
4 Inch Air-Operated Spray Valve Bonnet	200	9.3	50	50	4	190

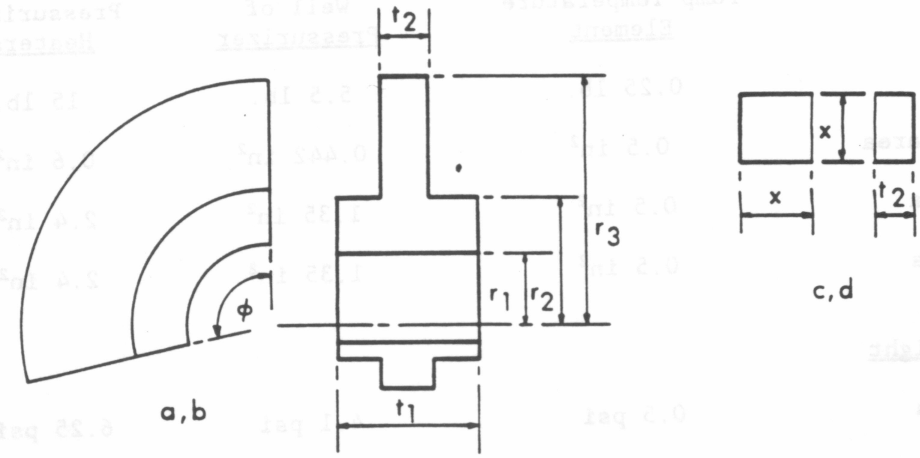
TABLE 3.5-4 PIPING TEMPERATURE ELEMENT ASSEMBLY - MISSILE CHARACTERISTICS

1. For a tear around the weld between the boss and the pipe:		
Characteristics	"Without well"	"With well"
Flow Discharge Area	0.11 in^2	0.60 in^2
Thrust Area	7.1 in^2	9.6 in^2
Missile Weight	11.0 lb.	15.2 lb.
Area of Impact	3.14 in^2	3.14 in^2
Missile Weight		
Impact Area	3.5 psi	4.84 psi
Velocity	20 ft/sec	120 ft/sec
2. For a tear at the junction between the temperature element assembly and the boss for the "without well" element and at the junction between the boss and the well for the "with well" element.		
Characteristics	"Without well"	"With well"
Flow Discharge Area	0.11 in^2	0.60 in^2
Thrust Area	3.14 in^2	3.14 in^2
Missile Weight	4.0 lb.	6.1 lb.
Area of Impact	3.14 in^2	3.14 in^2
Missile Weight		
Impact Area	1.27 psi	1.94 psi
Velocity	75 ft/sec	120 ft/sec

TABLE 3.5-5 CHARACTERISTICS OF OTHER MISSILES POSTULATED WITHIN REACTOR CONTAINMENT

	Reactor Coolant Pump Temperature Element	Instrument Well of Pressurizer	Pressurizer Heaters
Weight	0.25 lb.	5.5 lb.	15 lb.
Discharge area	0.5 in^2	0.442 in^2	0.6 in^2
Thrust area	0.5 in^2	1.35 in^2	2.4 in^2
Impact area	0.5 in^2	1.35 in^2	2.4 in^2
Missile Weight			
Impact Area	0.5 psi	4.1 psi	6.25 psi
Velocity	260 ft/sec	100 ft/sec	55 ft/sec

TABLE 3.5-6 **HYPOTHETICAL MISSILE DIMENSIONS**



MISSILE TYPE a : WHEEL SHAPE, $\phi = 120°$

MISSILE TYPE b : WHEEL SHAPE, $\phi = 60°$

MISSILE TYPE c : x_1 SQUARE, t_2 THICK

MISSILE TYPE d : x_2 SQUARE, t_2 THICK

WHEEL GROUP	r_1 (in)	r_2 (in)	r_3 (in)	t_1 (in)	t_2 (in)	x_1 (in)	x_2 (in)
1-3	20	27	48	9	3	19	11
4-6	18	27	47	12	5	20	10
7	17	28	45	27	12	20	8

TABLE 3.5-7 **HYPOTHETICAL MISSILE DATA**

Missile	**No. of**	**Type of**	**Wheel Group 1-3**			**Wheel Group 4-6**			**Wheel Group 7**		
Type	**Missiles**	**Overspeed (%)**	**Weight (lb.)**	**Energy (106ft lb.)**	**Velocity (ft/sec)**	**Weight (lb.)**	**Energy (106ft lb.)**	**Velocity (ft/sec)**	**Weight (lb.)**	**Energy (106ft lb.)**	**Velocity (ft/sec)**
a	2	180 120	2000	0-8	0-510	4000	0-17	0-520	8200	26-53 10-22	450-650 280-420
b	1	180 120	1000	0-8	0-720	2000	0-16	0-720	4100	0-38 0-18	0-770 0-530
c	3	180 120	300	0-5	0-1040	600	0-8	0-930	1400	0-16 0-8	0-860 0-610
d	10	180 120	100	0-2	0-1130	150	0-2	0-930	200	0-3 0-2	0-980 0-800

	Wheel Group 1-3	Wheel Group 4-6	Wheel Group 7
Missile Exit Angle[a]	-5 to + 5 degrees	-5 to + 5 degrees	0 to 25 degrees

a - From Reference 4

TABLE 3.5-8 **DATA ON SAFETY-RELATED STRUCTURES**

UNIT	BUILDING NO.	BUILDING NAME	WALL THICKNESS (ft)	ROOF THICKNESS (ft)	EFFECTIVE ELEVATION MINIMUM	TARGET (ft) MAXIMUM	SIZE	CONCRETE STRENGTH (psi)
1	1	Containment	4.5	-	20	119	74.25' radius	3000
1	2	Containment	3.5	3.5	119	183	74.25' radius	3000
1	3	Emergency Feedwater Pump	2.0	2.0	47	47	35'x82'	3000
1	4	Fuel Storage Building	2.0	2.0	20	84	98'x107.5'	3000
1	5	Primary, Auxiliary Bldg.	2.0	2.0	20	100	63.5'x145'	3000
1	6	Control Building	2.0	2.0	20	98	90'x138'	3000
1	7	Diesel Generator Bldg.	2.0	2.0	20	78.5	90'x93'	3000
1	8	Condensate Storage Tank	2.0	0.0	20	47.5	23.5' radius	3000
1	9	Equipment Vaults	2.5	2.5	2.5	53	41'x41.5'	3000
1	10	Service Water Pumphouse	2.0	2.0	20	50.25	60'x159'	3000

TABLE 3.5-9 PROBABILITY OF DAMAGE BY A LOW TRAJECTORY MISSILE STRIKE ON BUILDINGS IN DIRECT ZONE

Building	PROBABILITY OF DAMAGE PER TURBINE FAILURE (P2P3) ($x10^{-3}$ events/turbine failure)						Total Prob. of Damage (P1P2P3) ($x10^{-7}$ events/yr)
	120% Overspeed P1 = $1.89x10^{-6}$/year			180% Overspeed P1 = $1.89x10^{-6}$/year			
	Turbine 1	Turbine 2	Total	Turbine 1	Turbine 2	Total	
Containment	-	-	-	-	-	-	-
Control Building	-	-	-	-	-	-	-
Diesel Generator Bldg.	0.86	-	0.86	1.00	-	1.00	0.037
Condensate Storage Tank	-	-	-	-	-	-	-
Emerg. Feedwater Pump Bldg.	-	-	-	-	-	-	-
Total	0.86	-	0.86	1.00	-	1.00	0.037
Service Water Pumphouse	-	-	-	-	-	-	-

TABLE 3.5-10 PROBABILITY OF A HIGH TRAJECTORY MISSILE STRIKE - ROOF HITS ONLY

	PROBABILITY OF DAMAGE PER TURBINE FAILURE (P2) ($x10^{-3}$ events/turbine failure)						Total Prob. of Strike (P1P2) ($x10^{-7}$ events/yr)
	120% Overspeed P1 = $5.41x10^{-5}$/year			180% Overspeed P1 = $5.41x10^{-5}$/year			
Building	**Turbine 1**	**Turbine 2**	**Total**	**Turbine 1**	**Turbine 2**	**Total**	
Containment	.045	-	.045	.576	-	.576	0.337
Control Building	.032	-	.032	.414	-	.414	0.242
Diesel Generator Bldg.	.026	-	.026	.273	-	.273	0.162
Emerg. Feedwater Pump Bldg.	.007	-	.007	.094	-	.094	0.056
Equipment Vault	.004	-	.004	.055	-	.055	0.033
Primary Auxiliary Bldg.	.023	-	.023	.258	-	.258	0.153
Fuel Storage Building	.026	-	.026	.217	-	.217	0.133
Service Water Pumphouse	.023	-	.023	.194	-	.194	0.118

TABLE 3.5-11 TORNADO-GENERATED MISSILES AND VELOCITIES

MISSILE, DESCRIPTION	FRACTION OF TOTAL TORNADO VELOCITY[1]	HORIZONTAL VELOCITY IN REGION I (fps)
A. Wood plank, 4 in. x 12 in. x 12 ft weight 200 lb.	0.8	422
B. Steel pipe, 3 in. diameter, schedule 40 10 ft long, weight 78 lb.	0.4	211
C. Steel rod, 1 in. diameter x 3 ft long weight 8 lb.	0.6	317
D. Steel pipe, 6 in. diameter, schedule 40 15 ft long, weight 285 lb.	0.4	211
E. Steel pipe, 12 in. diameter, schedule 40 15 ft long, weight 743 lb.	0.4	211
F. Utility pole, 13½ in. diameter, 35 ft long, weight 1490 lb.	0.4	211
G. Automobile, frontal area 20 ft^2 weight 4000 lb.	0.2	106

[1] The maximum wind speed in Region is 360 MPH (528 fps) per Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants.

TABLE 3.5-12 STRUCTURES AND BARRIERS DESIGNED TO RESIST TORNADO-GENERATED MISSILES*

Containment Structure	Main Steam and Feedwater Pipe Chase (E&W)
Containment Enclosure Ventilation Area	Electrical Tunnel
Containment Equipment Hatch Missile Shield	Personnel Hatch Area
Control and Diesel Generator Building	Pre-Action Valve Building
Control Room Makeup Air Intake Structure	Primary Auxiliary Building
Dikes and Foundation for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank, walls E.7-D.1 @ Col. line 0.5 and E-5 @ Col. line 4.5	RHR and Containment Spray Equipment Vault
East Penetration Area	Service Water Pumphouse and external Barrier 1 Missile Barrier, including Electrical Room
Emergency Feedwater Pump Building	Waste Processing Building, external walls and roof below Elev. 53'-0" and including the following areas above Elev. 53'-0":
Enclosure for Condensate Water Storage Tank, vertical walls	Betw. Col. 1 & 2, Col's A to D to Elev. 86'-0"
Fuel Storage Building, except rolling steel door in external wall on Col. line A	Betw. Col. 4.9 & C, Col's B to E, to Elev. 86'-0"
Valve Pit Areas of the Intake and Discharge Transition Structure	Pipe Tunnel between Tank Farm and PAB
	Safety-Related Electrical Manholes

* Except where noted, all structures completely enclose the equipment housed therein.

Refer to Reference 20 for an evaluation of a missile entering a safety-related structure which resulted in a probability of about 10^{-6} per year.

TABLE 3.5-13 **AIRCRAFT ACCIDENT PROBABILITY**

Type Aircraft	N	Area(mi)^2	R Acc/Hr	Exp Hr/(mi)^2	P(yr)$^{-1}$
FB-111A	2,340	.0029	2.41×10^{-5}	4.61×10^{-4}	7.54×10^{-8}
FB-111A*	12	.005	2.41×10^{-5}	4.61×10^{-4}	6.67×10^{-10}
KC-135	2,903	.005	2.48×10^{-6}	3.89×10^{-4}	1.40×10^{-8}
C-130	384	.005	2.2×10^{-6}	3.89×10^{-4}	1.64×10^{-9}
Other**	459	.005	5.0×10^{-6}	3.89×10^{-4}	4.46×10^{-9}
Commercial	46,720	.005	3.89×10^{-7}	1.04×10^{-5}	9.45×10^{-10}

Total Aircraft Crash Probability = 9.71×10^{-8}

* Adjustment for FB-111A that weigh more than 81,800 pounds.

**Other includes C-123, F4, C5A, P3, B57, B52, and C-141 aircraft.
It also includes an additional 200 aircraft that may overfly the area (Subsection 2.2.2.5).

TABLE 3.6(B)-1 ESSENTIAL COMPONENTS LOCATED OUTSIDE CONTAINMENT POTENTIALLY SUSCEPTIBLE TO EFFECTS OF PIPING FAILURE

Component	Component Number	Location
Residual Heat Removal Pumps	RH-P-8A	Primary Auxiliary Building
	RH-P-8B	Primary Auxiliary Building
Safety Injection Pumps	SI-P-6A	Primary Auxiliary Building
	SI-P-6B	Primary Auxiliary Building
Chemical and Volume Control Charging Pumps	CS-P-2A	Primary Auxiliary Building
	CS-P-2B	Primary Auxiliary Building
Primary Component Cooling Water Pumps	CC-P-11A	Primary Auxiliary Building
	CC-P-11B	Primary Auxiliary Building
	CC-P-11C	Primary Auxiliary Building
	CC-P-11D	Primary Auxiliary Building
Emergency Feedwater Pumps	FW-P-37A	Emergency Feedwater Pump-house
	FW-P-37B	Emergency Feedwater Pump- house
Condensate Storage Tank	CO-TK-25	Yard

TABLE 3.6(B)-2 **HIGH ENERGY LINES**

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
1-1	1	Primary Loop	29	X		RC-20841
1-2	1	Primary Loop	31	X		RC-20841
2-1	1	Primary Loop	31	X		RC-20841
3-1	1	Primary Loop	27½	X		RC-20842
4-1	1	Primary Loop	29	X		RC-20842
4-2	1	Primary Loop	31	X		RC-20842
5-1	1	Primary Loop	31	X		RC-20842
6-1	1	Primary Loop	27½	X		RC-20842
7-1	1	Primary Loop	29	X		RC-20843
7-2	1	Primary Loop	31	X		RC-20843
8-1	1	Primary Loop	31	X		RC-20843
9-1	1	Primary Loop	27½	X		RC-20843
10-1	1	Primary Loop	29	X		RC-20844
10-2	1	Primary Loop	31	X		RC-20844
11-1	1	Primary Loop	31	X		RC-20844
12-1	1	Primary Loop	27½	X		RC-20844
13-1	1	Reactor Coolant	12	X		RC-20841
21-1	1	Reactor Coolant	4	X		RC-20841
48-1	1	Reactor Coolant	4	X		RC-20843, 20846
48-2	1	Reactor Coolant	6	X		RC-20846
48-3	1	Reactor Coolant	4	X		RC-20846
49-1	1	Reactor Coolant	14	X		RC-20843, 20846
58-1	1	Reactor Coolant	12	X		RC-20844
74-1	1	Reactor Coolant	6	X		RC-20846
75-1	1	Reactor Coolant	6	X		RC-20846
76-1	1	Reactor Coolant	6	X		RC-20846
80-1	1	Reactor Coolant	6	X		RC-20846
80-2	1	Reactor Coolant	3	X		RC-20846

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
80-6	1	Reactor Coolant	3	X		RC-20846
80-15	1	Reactor Coolant	6	X		RC-20846
93-1	1	Reactor Coolant	2	X		RC-20841
94-1	1	Reactor Coolant	2	X		RC-20842
96-1	1	Reactor Coolant	2	X		RC-20843
97-1	1	Reactor Coolant	3	X		RC-20843
97-2	2	Reactor Coolant	3	X		RC-20843, CS-20722
98-1	1	Reactor Coolant	2	X		RC-20844
155-5	1	Residual Heat Removal	6	X		RH-20662, SI-20450
155-17	1	Residual Heat Removal	10	X		SI-20450
158-3	2	Residual Heat Removal	8	X		RH-20663
158-4	2	Residual Heat Removal	6	X		RH-20663
158-5	1	Residual Heat Removal	6	X		RH-20663, SI-20450
158-17	1	Residual Heat Removal	10	X		SI-20450
160-6	1	Residual Heat Removal	6	X		RH-20663, RC-20844
160-17	1	Residual Heat	12	X		RC-20844
162-2	1	Residual Heat Removal	6	X		RH-20662, SI-20450
162-5	1	Residual Heat Removal	10	X		SI-20450
163-1	2	Residual Heat Removal	6	X		RH-20663
163-2	1	Residual Heat Removal	6	X		RH-20663, SI-20450
163-4	1	Residual Heat Removal	10	X		SI-20450
			Removal			
163-5	2	Residual Heat Removal	6	X		RH-20663
177-1	2	Residual Heat Removal	2	X		RH-20662, CS-20722
180-2	1	Residual Heat Removal	8	X		RH-20663

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
180-3	1	Residual Heat Removal	6	X		RH-20663, RC-20841
180-5	1	Residual Heat Removal	12	X		RC-20841
201-1	2	Safety Injection	10	X		SI-20450
201-2	1	Safety Injection	10	X		SI-20450, RC-20841
202-1	2	Safety Injection	10	X		SI-20450
202-2	1	Safety Injection	10	X		SI-20450, RC-20841
203-1	2	Safety Injection	10	X		SI-20450
203-2	1	Safety Injection	10	X		SI-20450, RC-20843
204-1	2	Safety Injection	10	X		SI-20450
204-2	1	Safety Injection	10	X		SI-20450, RC-20844
232-1	2	Safety Injection	2		X	SI-20450
232-2	2	Safety Injection	2		X	SI-20450
234-1	2	Safety Injection	2		X	SI-20450
234-2	2	Safety Injection	2		X	SI-20450
236-1	2	Safety Injection	2		X	SI-20450
236-2	2	Safety Injection	2		X	SI-20450
238-1	2	Safety Injection	2		X	SI-20450
238-2	2	Safety Injection	2		X	SI-20450
240-1	2	Safety Injection	2		X	SI-20450
240-2	2	Safety Injection	2		X	SI-20450
242-1	2	Safety Injection	2		X	SI-20450
242-2	2	Safety Injection	2		X	SI-20450
244-1	2	Safety Injection	2		X	SI-20450
244-2	2	Safety Injection	2		X	SI-20450
246-1	2	Safety Injection	2		X	SI-20450
246-2	2	Safety Injection	2		X	SI-20450

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
251-3	2	Safety Injection	4	X		SI-20446
251-5	1	Safety Injection	2	X		SI-20446
251-6	1	Safety Injection	3	X		SI-20446
251-7	1	Safety Injection	6	X		SI-20446, RC-20843
256-3	2	Safety Injection	2	X		SI-20446
256-4	1	Safety Injection	2	X		SI-20446, RH-20663
258-1	2	Safety Injection	2	X		SI-20446
258-2	1	Safety Injection	2	X		SI-20446, RH-20663
259-3	1	Safety Injection	2	X		SI-20446, RH-20662
260-2	1	Safety Injection	2	X		SI-20446, RH-20662
261-2	1	Safety Injection	2	X		SI-20448
261-3	1	Safety Injection	3	X		SI-20448
261-4	1	Safety Injection	6	X		SI-20446, RC-20842
270-2	1	Safety Injection	2	X		SI-20446, RH-20663
272-2	2	Safety Injection	4	X		SI-20447
272-3	2	Safety Injection	3	X		SI-20447
272-4	1	Safety Injection	3	X		SI-20447
272-5	1	Safety Injection	1½	X		SI-20447, RC-20841
272-9	2	Safety Injection	4	X		SI-20447
273-1	1	Safety Injection	1½	X		SI-20447
273-5	1	Safety Injection	3	X		SI-20447
274-1	1	Safety Injection	1½	X		SI-20447, RC-20844
274-5	1	Safety Injection	3	X		SI-20447

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
275-4	1	Safety Injection	1½	X		SI-20447
275-6	1	Safety Injection	3	X		SI-20447
318-1	2	Chem & Vol Control	3	X		CS-20722
318-2	2	Chem & Vol Control	2	X		CS-20722
318-9	2	Chem & Vol Control	2	X		CS-20722
324-1	2	Chem & Vol Control	3	X		CS-20722, CS-20726
325-1	2	Chem & Vol Control	2	X		CS-20726
326-1	2	Chem & Vol Control	2	X		CS-20726
327-1	2	Chem & Vol Control	3	X		CS-20726
327-2	2	Chem & Vol Control	2	X		CS-20726
328-1	2	Chem & Vol Control	2	X		CS-20726
328-2	2	Chem & Vol Control	3	X		CS-20726
328-3	2	Chem & Vol Control	2	X		CS-20726
328-6	1	Chem & Vol Control	2	X		CS-20726
328-7	1	Chem & Vol Control	1½	X		CS-20726
329-1	2	Chem & Vol Control	2	X		CS-20726
329-4	2	Chem & Vol Control	2	X		CS-20726
329-5	1	Chem & Vol Control	1½	X		CS-20726
330-1	2	Chem & Vol Control	2	X		CS-20726
330-4	1	Chem & Vol Control	2	X		CS-20726
330-5	1	Chem & Vol Control	1½	X		CS-20726
331-1	2	Chem & Vol Control	2	X		CS-20726
331-4	1	Chem & Vol Control	2	X		CS-20726
331-5	1	Chem & Vol Control	1½	X		CS-20726
348-1	NNS	Chem & Vol Control	3		X	CS-20724
354-1	NNS	Chem & Vol Control	3		X	CS-20724
355-1	2	Chem & Vol Control	3	X		CS-20722, CS-20725
355-6	2	Chem & Vol Control	3	X		CS-20722

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
356-1	2	Chem & Vol Control	4	X		CS-20725
356-2	2	Chem & Vol Control	3	X		CS-20725
358-1	2	Chem & Vol Control	2	X		CS-20725
358-2	2	Chem & Vol Control	3	X		CS-20725
358-3	2	Chem & Vol Control	2		X	CS-20725
360-1	2	Chem & Vol Control	3	X		CS-20722
360-2	2	Chem & Vol Control	2	X		CS-20722
360-3	2	Chem & Vol Control	2	X		CS-20722
360-4	2	Chem & Vol Control	3	X		CS-20722
360-5	2	Chem & Vol Control	2	X		CS-20722
360-6	2	Chem & Vol Control	2	X		CS-20722
360-7	2	Chem & Vol Control	3	X		CS-20722
360-8	2	Chem & Vol Control	4	X		CS-20722
360-9	2	Chem & Vol Control	3	X		CS-20722
360-25	2	Chem & Vol Control	3	X		CS-20722
360-26	2	Chem & Vol Control	3	X		CS-20722
361-1	2	Chem & Vol Control	2	X		CS-20722
362-1	2	Chem & Vol Control	4	X		CS-20725
363-1	2	Chem & Vol Control	2	X		CS-20725
363-2	2	Chem & Vol Control	3	X		CS-20725
363-3	2	Chem & Vol Control	2	X		CS-20725
364-1	2	Chem & Vol Control	4	X		CS-20725
364-2	2	Chem & Vol Control	3	X		CS-20725
364-9	2	Chem & Vol Control	1½	X		CS-20725
365-1	2	Chem & Vol Control	2	X		CS-20725
365-2	1	Chem & Vol Control	2	X		CS-20722, RC-20846
366-1	2	Chem & Vol Control	3	X		CS-20722
366-2	1	Chem & Vol Control	3	X		CS-20722, RC-20841

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
367-1	2	Chem & Vol Control	2	X		CS-20725
368-1	2	Chem & Vol Control	3	X		CS-20722
368-2	1	Chem & Vol Control	3	X		CS-20722
374-1	2	Chem & Vol Control	4	X		CS-20725
376-1	NNS	Chem & Vol Control	3		X	CS-20724
431-1	2	Chem & Vol Control	3	X		CS-20727, CS-20722
431-2	2	Chem & Vol Control	4	X		CS-20727
432-1	2	Chem & Vol Control	4	X		CS-20727
432-2	2	Chem & Vol Control	3	X		CS-20727,
511-9	3	Primary Component Cooling	3/4	X		CS-20722
525-1	NNS	Chem & Vol Control	4		X	CS-20724
525-2	NNS	Chem & Vol Control	3		X	CS-20724
525-3	NNS	Chem & Vol Control	3		X	CS-20724
526-1	NNS	Chem & Vol Control	2		X	CS-20724
526-2	NNS	Chem & Vol Control	2		X	CS-20724
526-8	NNS	Chem & Vol Control	1½		X	CS-20724
526-9	NNS	Chem & Vol Control	1½		X	CS-20724
526-10	NNS	Chem & Vol Control	3		X	CS-20724
526-13	NNS	Chem & Vol Control	2½		X	CS-20724
534-1	NNS	Chem & Vol Control	3		X	CS-20724
541-3	3	Primary Component Cooling	3/4	X		CS-20722
541-5	3	Primary Component Cooling	3/4	X		CS-20722
816-1	3	Primary Component Cooling	6	X		CS-20206
816-5	3	Primary Component Cooling	8	X		CS-20206
816-6	3	Primary Component Cooling	2	X		CS-20206
816-10	3	Primary Component Cooling	3	X		CS-20206
816-11	3	Primary Component Cooling	6	X		CS-20206
816-12	3	Primary Component Cooling	3/4	X		CS-20206
1301-1	2	Steam Gen Blowdown	2	X		RC-20841

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
1301-2	2	Steam Gen Blowdown	3	X		RC-20841, SB-20626
1301-3	2	Steam Gen Blowdown	2	X		SB-20626
1301-4	2	Steam Gen Blowdown	3	X		SB-20626
1301-5	NNS	Steam Gen Blowdown	3		X	SB-20626
1301-16	NNS	Steam Gen Blowdown	3		X	SB-20626
1301-22	NNS	Steam Gen Blowdown	1½		X	SB-20626
1301-23	NNS	Steam Gen Blowdown	4		X	SB-20626
1303-1	2	Steam Gen Blowdown	2	X		RC-20841
1304-1	2	Steam Gen Blowdown	2	X		RC-20842
1304-2	2	Steam Gen Blowdown	3	X		RC-20842, SB-20626
1304-3	2	Steam Gen Blowdown	2	X		SB-20626
1304-4	2	Steam Gen Blowdown	3	X		SB-20626
1304-5	NNS	Steam Gen Blowdown	3		X	SB-20626
1304-17	NNS	Steam Gen Blowdown	3		X	SB-20626
1304-24	NNS	Steam Gen Blowdown	1½		X	SB-20626
1304-25	NNS	Steam Gen Blowdown	4		X	SB-20626
1306-2	2	Steam Gen Blowdown	2	X		RC-20842
1307-1	2	Steam Gen Blowdown	2	X		RC-20843
1307-2	2	Steam Gen Blowdown	3	X		RC-20843, SB-20626
1307-3	2	Steam Gen Blowdown	2	X		SB-20626
1307-4	2	Steam Gen Blowdown	3	X		SB-20626
1307-5	NNS	Steam Gen Blowdown	3		X	SB-20626
1307-17	NNS	Steam Gen Blowdown	3		X	SB-20626
1307-22	NNS	Steam Gen Blowdown	1½		X	SB-20626
1307-23	NNS	Steam Gen Blowdown	4		X	SB-20626
1309-2	2	Steam Gen Blowdown	2	X		RC-20843
1310-1	2	Steam Gen Blowdown	2	X		RC-20844

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
1310-2	2	Steam Gen Blowdown	3	X		RC-20844, SB-20626
1310-3	2	Steam Gen Blowdown	2	X		SB-20626
1310-4	2	Steam Gen Blowdown	3	X		SB-20626
1310-5	NNS	Steam Gen Blowdown	3		X	SB-20626
1310-16	NNS	Steam Gen Blowdown	3		X	SB-20626
1310-22	NNS	Steam Gen Blowdown	1½		X	SB-20626
1310-23	NNS	Steam Gen Blowdown	4		X	SB-20626
1312-2	2	Steam Gen Blowdown	2	X		RC-20844
1317-1	NNS	Steam Gen Blowdown	10		X	SB-20626
1317-7	NNS	Steam Gen Blowdown	14		X	SB-20626
1319-1	NNS	Steam Gen Blowdown	10		X	SB-20626
1320-1	NNS	Steam Gen Blowdown	8		X	SB-20626
1320-2	NNS	Steam Gen Blowdown	3		X	SB-20626
1320-8	NNS	Steam Gen Blowdown	4		X	SB-20626
1321-1	NNS	Steam Gen Blowdown	3		X	SB-20626
1321-4	NNS	Steam Gen Blowdown	4		X	SB-20626
1350-3	NNS	Steam Gen Blowdown	10		X	SB-20626
2302-2	NNS	Aux Steam	8		X	AS-20570, AS-20569
2302-5	NNS	Aux Steam	4		X	AS-20570
2302-8	NNS	Aux Steam	4		X	AS-20570
2302-14	NNS	Aux Steam	2		X	AS-20571
2302-16	NNS	Aux Steam	2		X	AS-20571
2302-19	NNS	Aux Steam	4		X	AS-20570
2302-30	NNS	Aux Steam	4		X	AS-20571
2302-32	NNS	Aux Steam	8		X	AS-20570, AS-20571
2303-1	NNS	Aux Steam	6		X	AS-20570
2303-2	NNS	Aux Steam	4		X	AS-20570

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
2303-3	NNS	Aux Steam	4		X	AS-20570
2303-5	NNS	Aux Steam	2		X	AS-20570
2303-6	NNS	Aux Steam	3		X	AS-20570
2304-1	NNS	Aux Steam	3		X	AS-20570
2304-2	NNS	Aux Steam	1½		X	AS-20570
2304-3	NNS	Aux Steam	3		X	AS-20570
2306-1	NNS	Aux Steam	8		X	AS-20570
2306-2	NNS	Aux Steam	4		X	AS-20570
2306-4	NNS	Aux Steam	8		X	AS-20570
2306-5	NNS	Aux Steam	10		X	AS-20570
2309-1	NNS	Aux Steam	2		X	AS-20571
2309-2	NNS	Aux Steam	2		X	AS-20571
2339-1	NNS	Aux Steam	1½		X	AS-20571
2339-2	NNS	Aux Steam	1½		X	AS-20571
2341-1	NNS	Aux Steam	1½		X	AS-20571
2341-4	NNS	Aux Steam	2		X	AS-20571
2341-5	NNS	Aux Steam	1½		X	AS-20571
2364-1	NNS	Aux Steam	3		X	AS-20570
2365-6	NNS	Aux Steam	2		X	AS-20570
2366-1	NNS	Aux Steam	2		X	AS-20570
2401-1	NNS	Aux Steam Condensate	2½		X	ASC-20906
2401-2	NNS	Aux Steam Condensate	3		X	ASC-20906
2401-3	NNS	Aux Steam Condensate	2½		X	ASC-20906
2401-4	NNS	Aux Steam Condensate	1½		X	ASC-20906
2402-2	NNS	Aux Steam Condensate	2		X	ASC-20906
2402-3	NNS	Aux Steam Condensate	3		X	ASC-20906
2403-1	NNS	Aux Steam Condensate	1½		X	ASC-20906
2404-1	NNS	Aux Steam Condensate	1½		X	ASC-20906
2404-2	NNS	Aux Steam Condensate	3		X	ASC-20906

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
2404-3	NNS	Aux Steam Condensate	3		X	ASC-20906
2404-4	NNS	Aux Steam Condensate	6		X	ASC-20906
2404-5	NNS	Aux Steam Condensate	3		X	ASC-20906
2404-6	NNS	Aux Steam Condensate	1½		X	ASC-20906
2404-8	NNS	Aux Steam Condensate	3		X	ASC-20906
2406-1	NNS	Aux Steam Condensate	4		X	ASC-20906, ASC-20907
2406-2	NNS	Aux Steam Condensate	6		X	ASC-20906
2406-3	NNS	Aux Steam Condensate	4		X	ASC-20906
2406-4	NNS	Aux Steam Condensate	3		X	ASC-20906
2406-5	NNS	Aux Steam Condensate	3		X	ASC-20906
2407-4	NNS	Aux Steam Condensate	1½		X	ASC-20907
2409-4	NNS	Aux Steam Condensate	1½		X	ASC-20907
2410-1	NNS	Aux Steam Condensate	2		X	ASC-20906
2410-4	NNS	Aux Steam Condensate	1½		X	ASC-20906
2433-1	NNS	Aux Steam Condensate	2		X	ASC-20906
2433-2	NNS	Aux Steam Condensate	1¼		X	ASC-20906
2433-3	NNS	Aux Steam Condensate	1¼		X	ASC-20906
2433-4	NNS	Aux Steam Condensate	1¼		X	ASC-20906
2437-1	NNS	Aux Steam Condensate	2½		X	ASC-20906
2437-2	NNS	Aux Steam Condensate	3		X	ASC-20906
2437-3	NNS	Aux Steam Condensate	2½		X	ASC-20906
2437-4	NNS	Aux Steam Condensate	1½		X	ASC-20906
2438-2	NNS	Aux Steam Condensate	2		X	ASC-20906
2439-1	NNS	Aux Steam Condensate	1½		X	ASC-20906
2439-2	NNS	Aux Steam Condensate	1½		X	ASC-20906
2439-7	NNS	Aux Steam Condensate	1½		X	ASC-20907
2440-1	NNS	Aux Steam Condensate	1½		X	ASC-20906
2441-1	NNS	Aux Steam Condensate	1½		X	ASC-20907
2441-2	NNS	Aux Steam Condensate	1½		X	ASC-20907

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
2441-4	NNS	Aux Steam Condensate	2		X	ASC-20907
2441-7	NNS	Aux Steam Condensate	1½		X	ASC-20907
2442-1	NNS	Aux Steam Condensate	2		X	ASC-20906
2442-2	NNS	Aux Steam Condensate	2		X	ASC-20906
2450-1	NNS	Aux Steam Condensate	1¼		X	ASC-20906
2450-2	NNS	Aux Steam Condensate	1½		X	ASC-20906
2451-1	NNS	Aux Steam Condensate	1½		X	ASC-20906
4000-1	2	Mainsteam	32	X		MS-20580, MS-20583
4000-2	2	Mainsteam	30	X		MS-20580
4000-3	NNS	Mainsteam	30		X	MS-20583
4000-8	2	Mainsteam	6	X		MS-20580
4000-11	2	Mainsteam	6	X		MS-20580
4000-32	2	Mainsteam	6	X		MS-20580
4000-34	2	Mainsteam	6	X		MS-20580
4000-40	2	Mainsteam	30	X		MS-20580
4000-41	2	Mainsteam	30	X		MS-20580, MS-20583
4000-48	2	Mainsteam	2	X		MS-20580
4001-1	2	Mainsteam	32	X		MS-20581
4001-2	2	Mainsteam	30	X		MS-20581
4001-3	NNS	Mainsteam	30		X	MS-20583
4001-8	2	Mainsteam	6	X		MS-20581
4001-11	2	Mainsteam	6	X		MS-20581
4001-32	2	Mainsteam	6	X		MS-20581
4001-34	2	Mainsteam	6	X		MS-20581
4001-39	2	Mainsteam	30	X		MS-20581
4001-41	2	Mainsteam	30	X		MS-20580, MS-20583
4001-46	2	Mainsteam	2	X		MS-20581

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
4002-1	2	Mainsteam	32	X		MS-20581
4002-2	2	Mainsteam	30	X		MS-20581
4002-3	NNS	Mainsteam	30		X	MS-20583
4002-9	2	Mainsteam	6	X		MS-20581
4002-30	2	Mainsteam	6	X		MS-20581
4002-32	2	Mainsteam	6	X		MS-20581
4002-36	2	Mainsteam	30	X		MS-20581
4002-37	2	Mainsteam	30	X		MS-20581, MS-20583
4002-41	2	Mainsteam	2	X		MS-20581
4003-1	2	Mainsteam	32	X		MS-20580
4003-2	2	Mainsteam	30	X		MS-20580
4003-3	NNS	Mainsteam	30		X	MS-20580
4003-9	2	Mainsteam	6	X		MS-20580
4003-30	2	Mainsteam	6	X		MS-20580
4003-32	2	Mainsteam	6	X		MS-20580
4003-36	2	Mainsteam	30	X		MS-20580
4003-37	2	Mainsteam	30	X		MS-20580, MS-20583
4003-41	2	Mainsteam	2	X		MS-20580
4004-1	NNS	Mainsteam	48		X	MS-20583
4005-1	NNS	Mainsteam	24		X	MS-20583, MS-20585
4006-1	NNS	Mainsteam	30		X	MS-20583
4006-2	NNS	Mainsteam	6		X	MS-20583
4007-1	NNS	Mainsteam	30		X	MS-20583
4007-2	NNS	Mainsteam	6		X	MS-20583
4008-1	NNS	Mainsteam	30		X	MS-20583
4008-2	NNS	Mainsteam	6		X	MS-20583
4009-1	NNS	Mainsteam	30		X	MS-20583

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
4009-2	NNS	Mainsteam	6		X	MS-20583
4010-1	NNS	Mainsteam	24		X	MS-20583, MS-20585
4010-33	NNS	Mainsteam	6		X	MS-20583
4366-1	3	Diesel Generator - Air	2	X		DG-20460
4367-1	3	Diesel Generator - Air	2	X		DG-20460
4368-1	3	Diesel Generator - Air	2	X		DG-20465
4369-1	3	Diesel Generator - Air	2	X		DG-20465
4454-1	NNS	Mainsteam	2		X	MS-20587
4510-3	NNS	Mainsteam Drain	2		X	MS-20587
4511-3	NNS	Mainsteam Drain	2		X	MS-20587
4511-5	NSS	Mainsteam Drain	2		X	MS-20587
4513-3	NNS	Mainsteam Drain	2		X	MS-20587
4513-4	NNS	Mainsteam Drain	2		X	MS-20583
4515-3	NNS	Mainsteam Drain	2		X	MS-20587
4515-4	NSS	Mainsteam Drain	2		X	MS-20583
4517-3	NNS	Mainsteam Drain	2		X	MS-20587
4517-4	NNS	Mainsteam Drain	2		X	MS-20583
4519-3	NNS	Mainsteam Drain	2		X	MS-20587
4519-4	NNS	Mainsteam Drain	2		X	MS-20583
4606-2	NNS	Feedwater	18		X	FW-20686
4606-3	2	Feedwater	18	X		FW-20686
4606-4	2	Feedwater	16	X		FW-20686
4606-12	2	Feedwater	2	X		FW-20686
4606-15	2	Feedwater	4	X		FW-20686, FW-20688
4607-2	NNS	Feedwater	18		X	FW-20686
4607-3	2	Feedwater	18	X		FW-20686
4607-4	2	Feedwater	16	X		FW-20686
4607-12	2	Feedwater	2	X		FW-20686

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
4607-15	2	Feedwater	4	X		FW-20686, FW-20688
4608-2	NNS	Feedwater	18		X	FW-20686
4608-3	2	Feedwater	18	X		FW-20686
4608-4	2	Feedwater	16	X		FW-20686
4608-12	2	Feedwater	2	X		FW-20686
4608-15	2	Feedwater	4	X		FW-20686, FW-20688
4609-2	NNS	Feedwater	18		X	FW-20686
4609-3	2	Feedwater	18	X		FW-20686
4609-4	2	Feedwater	16	X		FW-20686
4609-12	2	Feedwater	2	X		FW-20686
4609-15	2	Feedwater	4	x		FW-20686, FW-20688
4614-2	2	Feedwater	4	X		FW-20686
4615-2	2	Feedwater	4	X		FW-20686
4616-2	2	Feedwater	4	X		FW-20686
4617-2	2	Feedwater	4	X		FW-20686
5198-1	NNS	Aux Steam	12		X	AS-20569
5198-2	NNS	Aux Steam	2		X	AS-20569
5198-3	NNS	Aux Steam	2		X	AS-20569
5198-8	NNS	Aux Steam	12		X	AS-20569
5198-9	NNS	Aux Steam	2		X	AS-20569
5198-11	NNS	Aux Steam	2		X	AS-20569
5198-13	NNS	Aux Steam	1½		X	AS-20569
5198-18	3	Aux Steam	12	X		AS-20569
5198-20	3	Aux Steam	12	X		AS-20569
5198-21	NNS	Aux Steam	1½		X	AS-20569
5198-22	NNS	Aux Steam	12		X	AS-20569

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
5230-1	NNS	Aux Steam	6		X	ASC-20908, ASC-20909, ASC-20906
5231-1	NNS	Aux Steam Condensate	2		X	ASC-20908
8751-1	NNS	Aux Steam Heat	1¼		X	ASC-20908
8757-1	NNS	Aux Steam Condensate	3		X	ASC-20906
8757-2	NNS	Aux Steam Condensate	2		X	ASC-20906
8757-5	NNS	Aux Steam Heat	2½		X	ASC-20906
8767-1	NNS	Aux Steam Heat	1¼		X	ASC-20908
8863-1	NNS	Aux Steam Condensate	2		X	ASC-20906
8864-1	NNS	Aux Steam Condensate	2½		X	ASC-20906
8864-2	NNS	Aux Steam Condensate	2		X	ASC-20906
9000-1	NNS	Hot Water Supply	6		X	HW-20051
9001-1	NNS	Hot Water Supply	6		X	HW-20051
9001-2	NNS	Hot Water Supply	3		X	HW-20051
9002-1	NNS	Hot Water Supply	4		X	HW-20051
9003-1	NNS	Hot Water Supply	4		X	HW-20051
9004-1	NNS	Hot Water Supply	1¼		X	HW-20051
9006-1	NNS	Hot Water Supply	2		X	HW-20051
9007-1	NNS	Hot Water Supply	4		X	HW-20051
9008-1	NNS	Hot Water Supply	4		X	HW-20051
9009-1	NNS	Hot Water Supply	1¼		X	HW-20051
9011-1	NNS	Hot Water Supply	2		X	HW-20051
9012-1	NNS	Hot Water Supply	1½		X	HW-20051
9013-1	NNS	Hot Water Supply	2		X	HW-20051
9014-1	NNS	Hot Water Supply	1½		X	HW-20051
9015-1	NNS	Hot Water Supply	2		X	HW-20051
9022-1	NNS	Hot Water Supply	4		X	HW-20051
9023-1	NNS	Hot Water Supply	4		X	HW-20051
9024-1	NNS	Hot Water Supply	4		X	HW-20051

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
9025-1	NNS	Hot Water Supply	4		X	HW-20051
9026-1	NNS	Hot Water Supply	4		X	HW-20051
9027-1	NNS	Hot Water Supply	4		X	HW-20051
9030-1	NNS	Hot Water Supply	2		X	HW-20056
9030-2	NNS	Hot Water Supply	1¼		X	HW-20056
9042-1	NNS	Hot Water Supply	2		X	HW-20051
9043-1	NNS	Hot Water Supply	2		X	HW-20051
9044-1	NNS	Hot Water Supply	2		X	HW-20051
9045-1	NNS	Hot Water Supply	2		X	HW-20051
9050-1	NNS	Hot Water Supply	2		X	HW-20051
9051-1	NNS	Hot Water Supply	2		X	HW-20051
9052-1	NNS	Hot Water Supply	2		X	HW-20051
9053-1	NNS	Hot Water Supply	2		X	HW-20051
9054-1	NNS	Hot Water Supply	2		X	HW-20051
9072-1	NNS	Hot Water Supply	2		X	HW-20051
9200-1	NNS	Hot Water Return	6		X	HWR-20051
9201-1	NNS	Hot Water Return	4		X	HWR-20051
9202-1	NNS	Hot Water Return	4		X	HWR-20051
9208-1	NNS	Hot Water Return	4		X	HWR-20051
9209-1	NNS	Hot Water Return	2		X	HWR-20051
9210-1	NNS	Hot Water Return	2		X	HWR-20051
9211-1	NNS	Hot Water Return	2		X	HWR-20051
9212-1	NNS	Hot Water Return	2		X	HWR-20051
9213-1	NNS	Hot Water Return	2		X	HWR-20051
9214-1	NNS	Hot Water Return	4		X	HWR-20051
9215-1	NNS	Hot Water Return	2		X	HWR-20051
9216-1	NNS	Hot Water Return	2		X	HWR-20051
9217-1	NNS	Hot Water Return	2		X	HWR-20051
9218-1	NNS	Hot Water Return	2		X	HWR-20051

Line No.	Safety Class	Essential Function	Size	Yes	No	P&ID
9219-1	NNS	Hot Water Return	2		X	HWR-20051
9220-2	NNS	Hot Water Return	1¼		X	HWR-20056
9220-3	NNS	Hot Water Return	2		X	HWR-20051, HWR-20056
9221-3	NNS	Hot Water Return	1½		X	HWR-20056
9222-3	NNS	Hot Water Return	1½		X	HWR-20056
9223-2	NNS	Hot Water Return	1¼		X	HWR-20056
9223-3	NNS	Hot Water Return	1½		X	HWR-20056
9268-3	NNS	Hot Water Return	1½		X	HW-20056
9270-2	NNS	Hot Water Return	1½		X	HW-20056
9270-3	NNS	Hot Water Return	1¼		X	HW-20056
9271-2	NNS	Hot Water Return	1½		X	HW-20052
9271-6	NNS	Hot Water Return	1½		X	HW-20052
9272-3	NNS	Hot Water Return	1½		X	HW-20052
9273-3	NNS	Hot Water Return	1½		X	HW-20052
9274-3	NNS	Hot Water Return	1½		X	HW-20052
9275-3	NNS	Hot Water Return	1½		X	HW-20052
9826-1	NNS	Hot Water Supply	1½		X	HW-20056
9830-1	NNS	Hot Water Supply	1½		X	HW-20056
9831-1	NNS	Hot Water Supply	1½		X	HW-20056
9832-1	NNS	Hot Water Supply	1½		X	HW-20056
9833-1	NNS	Hot Water Supply	1½		X	HW-20056
9834-3	NNS	Hot Water Supply	1½		X	HW-20056
9835-1	NNS	Hot Water Supply	1½		X	HW-20056
9835-4	NNS	Hot Water Supply	1½		X	HW-20056
9836-3	NNS	Hot Water Supply	1½		X	HW-20056
9849-1	NNS	Hot Water Supply	1½		X	HW-20052
9849-7	NNS	Hot Water Supply	1½		X	HW-20052
9850-3	NNS	Hot Water Supply	1½		X	HW-20052
9851-3	NNS	Hot Water Supply	1½		X	HW-20052

TABLE 3.6(N)-1* **POSTULATED BREAK LOCATIONS FOR THE LOCA ANALYSIS OF THE PRIMARY COOLANT LOOP**

Location of Postulated Rupture	Type	Break Opening Area*
1. Residual Heat Removal (RHR) Line/Primary Coolant Loop Connection	Guillotine (viewed from the RHR line)	Cross-Sectional Flow Area of the RHR Line
2. Accumulator (ACC) Line/ Primary Coolant Loop Connection	Guillotine (viewed from the ACC line)	Cross-Sectional Flow Area of the ACC line
3. Pressurizer Surge (PS) Line/Primary Coolant Loop Connection	Guillotine (viewed from the PS line)	Cross-Sectional Flow Area of the PS line
* Refer to Figure 3.6(N)-2 for location of postulated breaks in reactor coolant loop.		
** Less break opening area will be used if justified by analysis, experiments or considerations of physical restraints such as concrete walls or structural steel.		

TABLE 3.7(B)-1 DAMPING VALUES*

	(Percent of Critical Damping)	
Structure or Component	**Operating Basis Earthquake**	**Safe Shutdown Earthquake**
Equipment and large-diameter piping systems, pipe diameter greater than 12 in.	2	3
Small-diameter piping systems, diameter equal to or less than 12 in.	1	2
Welded steel structures	2	4
Bolted steel structures	4	7
Prestressed concrete structures	2	5
Reinforced concrete structures	4	7

Reference

Newark, N.M., John A. Blume, and Kanwar K. Kapur, "Design Response Spectra for Nuclear Power Plants," ASCE Structural Engineering Meeting, San Francisco, April 1973.

* For seismic piping analysis, the values in Figure 3.7(B)-38 may be used as an alternative.

TABLE 3.7(B)-2 CONTAINMENT SHELL - SEISMIC MODEL

a. Beam Section Properties			
Beam No.	**Area ($in^2 x 10^{10}$)**	**Moment of Inertia ($in^4 x 10^{10}$)**	**Shear Shape* Factor**
1	2.272	1.462	0.53
2	2.272	3.342	0.53
3	2.272	5.402	0.53
4	2.272	7.035	0.53
5	2.272	7.926	0.53
6	2.272	8.371	0.53
7	2.942	11.067	0.53
8	2.942	11.067	0.53
9	2.942	11.067	0.53
10	2.942	11.067	0.53
11	2.942	11.067	0.53
12	2.942	11.067	0.53
13	2.942	11.067	0.53
14	2.942	11.067	0.53
15	2.942	11.067	0.53
16	2.942	11.067	0.53
17	2.942	11.067	0.53
18	2.942	11.067	0.53
19	2.942	11.067	0.53
20	2.942	11.067	0.53
* Area x shear shape factor = Shear area			

b. Modal Weight & Weight Moment of Inertia		
Node	**W (Kipx10^3)**	**(K-in^{2}x10^8)**
1	1.166	0.761
2	2.332	2.483
3	3.129	6.437
4	3.395	9.158
5	2.864	9.423
6	2.864	10.262
7	2.624	13.581
8	4.776	18.102
9	5.167	19.605
10	4.071	15.416
11	4.424	18.781
12	5.872	22.337
13	4.893	18.589
14	3.915	14.800
15	3.367	12.715
16	4.697	17.859

TABLE 3.7(B)-3 CONTAINMENT INTERNALS MATHEMATICAL MODEL

a. Summary of Section Properties						
Beam No.	Area (in^2)	IX2 (in^4)	IX3 (in^4)	J (in^4)	SF2* 0.33	SF3 0.70
30,31	$3.523x10^5$	$0.171x10^{10}$	$3.406x10^{10}$	$5.818x10^{10}$	0.33	0.70
27	$3.598x10^5$	$0.348x10^{10}$	$0.517x10^{10}$	$6.556x10^{10}$	0.40	0.62
25	$4.093x10^5$	$0.380x10^{10}$	$0.549x10^{10}$	$6.407x10^{10}$	0.41	0.61
* SF2 = Shear shape factor for bending about X2(N-S) axis						
b. Summary of Nodal Weights						
Node	WX1	WX2,WX3	WX4			
P4	($lbx10^6$)	($lbx10^6$)	($lb\text{-}in^2x10^{11}$)			
29	6.061	9.711	23.581			
26	6.913	8.727	24.187			
23	6.780	6.780	11.694			

TABLE 3.7(B)-4 PRIMARY AUXILIARY BUILDING - SEISMIC MODEL

a. Beam Properties

Beam No.	Elev (ft)		Area (in^2)	Center of Rigidity (in)		Moment of Inertia (in^4)		Torsional Constant, J (in^4)	Shear Shape Factor*	
	From	To		Xcr	Ycr	Ix-x	Iy-y		SFX	SFY
2	108	81	61,632	1050.72	323.40	8.665×10^8	1.2825×10^9	6.3348×10^9	0.4673	0.5327
5	81	53	210,586	722.64	539.18	4.3234×10^9	1.9511×10^{10}	5.7646×10^{10}	0.4721	0.5279
8	53	25	194,256	757.20	483.12	4.0186×10^9	1.1467×10^{10}	6.7102×10^{10}	0.4694	0.5306
11	25	73	39,552	853.20	536.52	5.6858×10^9	1.7503×10^{10}	8.6490×10^{10}	0.6166	0.3834

* Shear area for bending about X-X = Area X SFX

Shear area for bending about Y-Y = Area X SFY

b. Weight, Weight Moment of Inertia, Center of Weight

								Mass Center
Node No.	Elev (ft)	Weight (10^6lbs)			Weight Moment of Inertia,	Mass Center (in)		
			IX	IY	IZ	XBAR	YBAR	
2	108	2.1897	100.166	116.580	213.590	1044.00	318.00	
5	81	9.4550	860.672	2481.696	3326.308	811.58	453.26	
8	53	11.0715	1039.794	3167.405	4185.158	741.79	501.62	
11	25	11.2919	996.351	3068.816	4048.544	851.51	487.97	

TABLE 3.7(B)-5 PRIMARY AUXILIARY BUILDING - SEISMIC ANALYSIS DYNAMIC PROPERTIES - COUPLED TORSION MODEL

Mode No.	Natural Frequency (Hz)	Generated Weight, (10^7 lbs)	Participation Factor		Remarks - Type of Mode
			X2 (N-S)	X3 (E-W)	
1	5.0775	0.7331	0.0093	-1.7114	Bending About X2 (N-S) Axis
2	7.0437	0.8780	1.6649	-0.0155	Bending About X3 (E-W) Axis
3	13.0131	5.1185	-0.0639	-.0639	Torsion
4	16.8425	0.4718	0.0834	-1.0227	Bending About X2 (N-S) Axis
5	19.0848	0.4287	-1.0773	-0.0534	Bending About X3 (E-W) Axis
6	26.5067	1.1298	-0.0082	0.4578	Bending About X2 (N-S) Axis
7	27.7903	0.9912	0.5293	0.0008	Bending About X3 (E-W) Axis
8	32.3146	101.8944	0.0055	0.0033	Torsion
9	35.9770	2.1381	0.2764	0.0142	Bending About X3 (E-W) Axis
10	37.3225	1.3661	-0.0126	0.5899	Bending About X2 (N-S) Axis
11	37.7250	16.0945	0.0011	0.0138	Torsion
12	48.1893	32.5880	0.0052	0.0051	Torsion

TABLE 3.7(B)-6 PRIMARY AUXILIARY BUILDING - VERTICAL SEISMIC ANALYSIS MODEL

a. Nodal Weights and Spring Constants

El. (ft)	Node No.	Wt. (Kips)	Frequency (Hz)	Constant (10^3 Kips/in)	Mode Classification
108	1	1655	21.599	-	1st Structural Frequency
	2	535	4.054	0.901	Local Frequency
81	3	7615	21.599	-	1st Structural Frequency
	4	519	7.184	2.749	Local Frequency
	5	964	10.225	10.449	Local Frequency
	6	228	12.654	3.787	Local Frequency
	7	129	24.220	7.403	Local Frequency
65.75	9	40	8.097	0.268	Local Frequency
53	10	9306	21.599	-	1st Structural Frequency
	11	150	7.596	0.885	Local Frequency
	12	1338	8.950	11.073	Local Frequency
	13	168	10.173	1.786	Local Frequency
	14	110	18.424	3.888	Local Frequency
25	15	9488	21.599	-	1st Structural Frequency
	16	385	7.497	2.213	Local Frequency
	17	740	8.885	5.989	Local Frequency
	18	249	11.591	3.423	Local Frequency
	19	430	17.032	12.846	Local Frequency
7	20	375	9.300	3.314	Local Frequency
	21	96.0	14.401	2.034	Local Frequency

Beam No.	Axial Area (in^2)
1	61632
2	210586
3	210586
4	194256
5	339552

TABLE 3.7(B)-7 **CONTROL AND DIESEL GENERATOR BUILDING SEISMIC ANALYSIS MODEL**

a. Summary of Nodal Weights

		NODE NO.	WEIGHT (lbs)	I-(lb-in^2) About X1 (Vert.)		
		3	7.9198×10^6	2.5171×10^{12}		
		6	5.6269×10^6	1.1731×10^{12}		
		9	8.6589×10^6	2.6786×10^{12}		
		12	7.0052×10^6	1.3945×10^{12}		
		15	7.5876×10^6	2.6709×10^{12}		

b. Summary of Beam Properties

			Area Moment of Inertia			
Beam No.	**Area (in^2)**	**Torsional Constant (J) (in^4)**	**About X2 IX2 (in^4)**	**About X3 IX3 (in^4)**	**SF2**	**SF3**
1	48744	6.524×10^9	2.767×10^{10}	2.355×10^9	0.370	0.520
2	51840	7.387×10^9	2.767×10^{10}	2.355×10^{10}	0.511	0.489
3	64800	1.100×10^{10}	8.887×10^9	2.355×10^9	0.609	0.391
4	64800	1.100×10^{10}	2.767×10^{10}	2.355×10^9	0.609	0.391
5	64800	1.100×10^{10}	2.767×10^{10}	2.355×10^9	0.609	0.391
6	64800	1.100×10^{10}	8.887×10^9	2.355×10^{10}	0.609	0.391
7	116640	9.314×10^9	2.767×10^{10}	4.710×10^9	0.565	0.435
8	101448	1.293×10^{10}	2.767×10^{10}	3.188×10^8	0.567	0.437
9	116640	9.320×10^9	2.767×10^{10}	4.710×10^9	0.565	0.435
10	116640	1.293×10^{10}	2.767×10^{10}	4.710×10^9	0.565	0.435

TABLE 3.7(B)-8 FUEL STORAGE BUILDING SEISMIC ANALYSIS MODEL

a. Element (Beam) Properties and Nodal Weights

Model	Beam No.	Area (in^2)	Torsional Constant (J) ($10^{10}in^4$)	Moment of Inertia ($10^{10}in^4$)		Shear Shape Factor For Bending	
				About X2	About X3	SF2	SF3
Model for N - S Motion (X2)	1	61776	2.111	.4066	.3203	.647	.523
	2	75456	1.940	.4066	.4805	.530	.491
	3	61776	1.976	.4066	.3203	.568	.435
	4	75456	1.940	.4066	.4805	.465	.491
	5	61776	2.428	.4066	.3203	.590	.364
	6	75456	1.985	.4066	.4805	.483	.573
	7	61776	2.428	.4066	.3203	.590	.364
	8	75456	1.985	.4066	.4805	.483	.573
	9	378300	6.849	1.186	1.171	.522	.634

b. Nodal Weight and Weight Moment of Inertia

Node No.	Translational Inertia (lbs) X2	Rotational Inertia (10^{11} lb-in^2) About X1
2	5051000	14.21
4	3216000	9.953
5	797000	.846
6	941000	2.709
7	1154000	2.493
8	648000	1.805
9	873000	2.047
12	1887000	5.683

c. Element (Beam) Properties and Nodal Weights							
Model	Beam No.	Area (in^2)	Torsional Constant (J) 10^{10} in^4	Moment of Inertia (10^{10} in^4)		Shear Shape Factor For Bending	
				About X2	About X3	SF_2	SF_3
Model for E – W (X3)	1	137232	4.051	8.132	8.008	.583	.486
	2	137232	3.916	8.132	8.008	.512	.422
	3	137232	4.232	8.132	8.008	.537	.484
	4	137232	4.232	8.132	8.008	.537	.484
	5	378300	6.849	1.186	1.171	.522	.634
	6	15329	-	.0003105	-	-	-
	7	15329	-	.0003105	-	-	-

d. Nodal Weights and Weights Moment of Inertia		
Node No.	Translational Inertia (lbs) X3	Rotational Inertia (10^{12} lb-in^2) About X1
2	4903000	1.379
5	3707000	1.270
6	843000	-
8	1859000	0.894
9	1344000	0.644
12	1772000	0.568

e. Nodal Weight and Beam Properties (Vertical Model)							
	Node No.	Nodal Weight (lbs)	Element No.	Axial Area (in^2)	Spring Constant (lbs/in)		
	1	850000	1	137232	1389000		
	2	1203000	2	137232	3718000		
	3	271000	3	137232	5049900		
	4	2727000	4	137232	-		
	5	563000	5	378300	920300		
	6	140000	-	-	1033000		
	7	3310000	-	-	-		
	8	2093000	-	-	-		
	9	1521000	-	-	-		
	10	1656000	-	-	-		
	11	178000	-	-	1473000		
	12	84000	-	-	1236000		
	13	430000	-	-	8610000		
	14	90000	-	-	3680000		

f. Nodal Weight and Beam Properties-Crane Girder System

		Node No.		**Nodal Weight**		**Element No**	**Axial Area (inches^2)**
		1		-		1	150
		2		7400		2	150
		3		7400		3	150
		4		6800		4	150
		5		5100		5	150
		6		34700		6	150
		7		34700		7	150
		8		3500		8	150
		9		37600		9	150
		10		34000		10	150
		11		5200		11	150
		12		9200		12	150
		13		9200		13	150
		14		-		14	40
		15		-		15	40
		16		7700		16	40
		17		7700		17	40
		18		-		18	40
		19		1600		19	40
		20		-		20	40
		21		-		21	40
						22	40
						23	11.8
						24	11.8
						25	11.8
						26	11.8
						27	40
						28	40

TABLE 3.7(B)-9 CONTAINMENT STRUCTURE SEISMIC ANALYSIS NATURAL FREQUENCIES, PERIODS AND MODE CLASSIFICATION

a. Shell

Mode Number	Natural Frequency (Hz)	Natural Period (sec.)	Mode Classification
1	4.17	.2398	1st Horizontal
2	11.78	.0849	1st Vertical
3	11.84	.0845	2nd Horizontal
4	22.01	.0454	1st Rotational
5	24.17	.0414	3rd Horizontal
6	33.15	.0302	2nd Vertical
7	34.81	.0287	Higher Modes
8	40.00	.0250	Higher Modes
9	45.88	.0218	Higher Modes
10	53.28	.0188	Higher Modes
11	53.62	.0186	Higher Modes
12	59.91	.0167	Higher Modes

b. Internals

Mode Number	Natural Frequency (Hz)	Natural Period (sec.)	Mode Classification
1	9.07	.1103	1st Horizontal (E-W)
2	11.90	.0840	1st Horizontal (N-S)
3	20.25	.0494	1st Torsion
4	28.37	.0353	2nd Horizontal (E-W)
5	40.09	.0249	2nd Horizontal (N-S)
6	44.88	.0223	1st Vertical
7	49.21	.0203	Higher Modes
8	73.40	.0136	Higher Modes

TABLE 3.7(B)-10 CONTAINMENT STRUCTURE SEISMIC ANALYSIS - PEAK NODAL ACCELERATIONS

a. Peak Nodal Accelerations Due to Horizontal SSE and OBE Excitations - Containment Shell

		Horiz. Nodal Accelerations (g)			
		Safe Shutdown Earthquake		Operating Basis Earthquake	
Mass Elevation (feet)	**Node Number**	**Spectrum Method**	**Time History Method**	**Spectrum Method**	**Time History Method**
189.0	1	.969	1.069	.619	.692
179.5	2	.926	-	.593	-
170.0	3	.879	.995	.563	.642
154.0	4	.798	.923	.513	.592
142.3	5	.741	.875	.476	.556
130.7	6	.687	-	.441	-
119.0	7	.638	.769	.409	.487
105.0	8	.588	.707	.376	.450
88.5	9	.533	.637	.339	.415
72.0	10	.479	.581	.304	.373
62.5	11	.448	.545	.283	.351
43.8	12	.382	-	.239	-
37.3	31	.354	.452	.222	.284
30.0	32	.326	.422	.203	.263
25.0	13	.308	-	.190	-
18.0	33	.275	.370	.170	.226
12.5	14	.252	.345	.154	.209
0.0	15	.250	.294	.130	.169
-9.0	16	.250	.272	.130	.146
-18.5	34	.250	.256	.130	.131

b. Peak Nodal Accelerations Due to Vertical SSE and OBE Excitations -Containment Shell					
		Vertical. Nodal Accelerations (g)			
		Safe Shutdown Earthquake		Operating Basis Earthquake	
Mass Elevation (feet)	**Node Number**	**Spectrum Method**	**Time History Method**	**Spectrum Method**	**Time History Method**
189.0	1	.644	.565	.390	.342
179.5	2	.641	-	.389	-
170.0	3	.634	.559	.385	.340
154.0	4	.614	.545	.374	.334
142.3	5	.594	.532	.362	.328
130.7	6	.570	-	.348	-
119.0	7	.543	.504	.331	.311
105.0	8	.514	.491	.313	.301
88.5	9	.473	.474	.288	.287
72.0	10	.427	.452	.259	.269
62.5	11	.397	.437	.241	.257
43.8	12	.334	-	.201	-
37.3	31	.308	.387	.185	.218
30.0	32	.281	.372	.168	.206
25.0	13	.262	-	.156	-
18.0	33	.233	.347	.138	.190
12.5	14	.211	.336	.125	.183
0.0	15	.167	.311	.091	.167
-9.0	16	.167	.293	.083	.155
-18.5	34	.167	.273	.083	.141

c. Peak Nodal Acceleration (g) - Containment Internals					
Earthquake Direction	**Node No.**		**Elevation (ft)**	**SSE 7% Damping Spectrum Method**	**OBE 4% Damping Spectrum Method**
	29		25'-0"	.249	.1246
(Vertical)	26		0'-0"	.167	.0834
X_1	23		-9'-0"	.167	.0830
	29		25'-0"	.617	.375
(N - S)	26		0'-0"	.307	.181
X_2	23		-9'-0"	.250	.130
	29		25'-0"	.726	.470
(E - W)	26		0'-0"	.322	.200
Y_3	23		-9'-0"	.250	.130

TABLE 3.7(B)-11 CONTAINMENT STRUCTURE SEISMIC ANALYSIS - PEAK NODAL DISPLACEMENTS

a. Peak Nodal Displacements Due to Horizontal SSE and OBE Excitations -Containment Shell

		Horizontal Nodal Displacement (inches)			
		Safe Shutdown Earthquake		Operating Basis Earthquake	
Mass Elevation (feet)	**Node Number**	**Spectrum Method**	**Time History Method**	**Spectrum Method**	**Time History Method**
189.0	1	.516	.618	.332	.393
179.5	2	.498	-	.321	-
170.0	3	.479	.573	.308	.365
154.0	4	.443	.531	.285	.338
142.3	5	.415	.498	.267	.317
130.7	6	.386	-	.248	-
119.0	7	.355	.427	.228	.271
105.0	8	.320	.385	.206	.244
88.5	9	.277	.334	.178	.212
72.0	10	.233	.282	.150	.179
62.5	11	.208	.252	.134	.160
43.8	12	.159	-	.103	-
37.3	31	.143	.174	.092	.110
30.0	32	.125	.152	.080	.097
25.0	13	.113	-	.072	-
18.0	33	.096	.118	.062	.074
12.5	14	.083	.102	.054	.065
0.0	15	.056	.069	.036	.044
-9.0	16	.038	.047	.024	.029
-18.5	34	.019	.024	.013	.015

b. Peak Nodal Displacements Due to Vertical SSE and OBE Excitations - Containment Shell					
		Vertical Nodal Displacements (inches)			
		Safe Shutdown Earthquake		Operating Basis Earthquake	
Mass Elevation (feet)	Node Number	Spectrum Method	Time History Method	Spectrum Method	Time History Method
189.0	1	.0444	.0434	.0271	.0262
179.5	2	.0443	-	.0270	-
170.0	3	.0439	.0430	.0268	.0259
154.0	4	.0428	.0419	.0261	.0253
142.3	5	.0416	.0408	.0254	.0247
130.7	6	.0400	-	.0244	-
119.0	7	.0381	.0377	.0233	.0228
105.0	8	.0360	.0357	.0220	.0216
88.5	9	.0330	.0329	.0202	.0199
72.0	10	.0295	.0297	.0180	.0179
62.5	11	.0273	.0276	.0167	.0166
43.8	12	.0224	-	.0137	-
37.3	31	.0206	.0212	.0126	.0128
30.0	32	.0185	.0192	.0113	.0115
25.0	13	.0171	-	.0105	-
18.0	33	.0150	.0158	.0092	.0094
12.5	14	.0134	.0142	.0082	.0084
0.0	15	.0096	.0102	.0058	.0061
-9.0	16	.0067	.0073	.0041	.0043
-18.5	34	.0037	.0040	.0022	.0024

c. Peak Nodal Displacement (inch) - Containment Internals

Earthquake Direction	Node No.	Elevation (ft)	SSE 7% Damping Spectrum Method	OBE 4% Damping Spectrum Method
	29	25'-0"	.001181	.000591
(Vertical)	26	0'-0"	.000780	.000390
X_1	23	-9'-0"	.000532	.000266
	29	25'-0"	.04206	.02579
(N - S)	26	0'-0"	.01900	.01157
X_2	23	-9'-0"	.01048	.00638
	29	25'-0"	.08614	.05538
(E - W)	26	0'-0"	.03169	.02035
X_3	23	-9'-0"	.01730	.01109

TABLE 3.7(B)-12 CONTAINMENT STRUCTURE SEISMIC ANALYSIS - PEAK FORCES

a. Peak Seismic Forces - Containment Shell

		OBE (.13g) 4% Damping			SSE(.25g) 7% Damping	
Beam No.	Axial Force (Kips)	Shear Force (Kips)	Overturning Moment (10^3 Kips-in)	Axial Force (Kips)	Shear Force (Kips)	Overturning Moment (10^3Kips-in)
1	455	722	100	751	1130	157
2	1363	2104	396	2247	3290	622
3	2567	3865	1281	4230	6038	2009
4	3833	5595	2262	6308	8728	3545
5	4866	6942	3428	8001	10819	5366
6	5856	8178	4772	9620	12732	7463
7	7047	9604	6631	11563	14941	10356
8	8527	11300	9162	13978	17565	14292
9	9994	12903	11992	16371	20053	18687
10	11027	13988	13783	18057	21741	21466
11	12067	15069	17312	19756	23426	26938
14	13206	16212	21112	21619	21515	32831
16	13935	16920	23744	22814	26329	36914
17	14393	17357	26378	23567	27020	41000
18	14675	17619	28294	24032	27437	43973
20	14953	17873	32744	24491	27843	50884

b. Peak Forces Due to Horizontal (N-S) Excitation - Containment Internals

Earthquake	Elevation	Beam No.	Shear Forces N-S Component (Kips)	Bending Moments E-W Component (Kips)
OBE	-26'	25	5748	2.80
	-9'	27	5092	1.64
	0'	30	3644	1.093
	12.5'	31	8644	0.54711
SSE	-26'	25	9458	4.597
	-9'	27	8356	2.690
	0'	30	5992	1.798
	12.5'	31	5992	0.898

c. Peak Forces Due to Horizontal (E-W) Excitation - Containment Internals

Earthquake	Elevation	Beam No.	Shear Forces N-S Component (Kips)	Bending Moments E-W Component (Kips)	Torque 10^6 Kip-in
OBE	-26'	25	6685	3.352	0.4487
	-9'	27	6000	2.000	0.4172
	0'	30	4525	1.357	0.2252
	12.5'	31	4525	0.6787	0.2252
SSE	-26'	25	10454	5.215	0.699
	-9'	27	9359	3.113	0.649
	0'	30	7052	2.116	0.353
	12.5	31	7052	1.058	0.353

d. Peak Axial Forces Due to Vertical SSE and OBE Excitations By Response Spectrum Method - Containment Internals

		Axial Forces (Kips)	
Beam No.	**Elevation**	**Safe Shutdown Earthquake**	**Operating Basis Earthquake**
25	-26'-0" to -9'-0"	3331	1665
27	-9'-0" to 0'-0"	2582	1290
30	0'-0" to 12'-6"	1510	755
31	12'-6" to 25'-0"	1510	755

TABLE 3.7(B)-13 PRIMARY AUXILIARY BUILDING SEISMIC ANALYSIS - PEAK NODAL

a. Peak Nodal Displacements and Accelerations due to X2 (N-S) and X3 (E-W) Motions ÷ (RSS Sum)

Direction of Response	Node No.	OBE Due to X2 Motion	OBE Due to X3 Motion	OBE SRSS	SSE Due to X2 Motion	SSE Due to X3 Motion	SSE SRSS
X2	2	0.1242	0.0027	0.1242	0.1931	0.0042	0.1931
(N-S)	5	0.0862	0.0013	0.0862	0.1339	0.0020	0.1339
	8	0.0508	0.0007	0.0508	0.0789	0.0011	0.0789
	11	0.0147	0.0002	9.0147	0.0229	0.0003	0.0229
X3	2	0.0019	0.2572	0.2572	0.0030	0.3997	0.3997
(E-W)	5	0.0013	0.1660	0.1660	0.0020	0.2579	0.2579
	8	0.0010	0.0832	0.0832	0.0015	0.1293	0.1293
	11	0.0002	0.0154	0.0154	0.0003	0.0239	0.0239
X2	2	0.6682	0.0253	0.6687	1.0593	0.0438	1.0602
(N-S)	5	0.4400	0.0059	0.4400	0.6861	0.0095	0.6862
	8	0.2705	0.0105	0.2707	0.4279	0.0181	0.4283
	11	0.1300	0.0046	0.1301	0.2500	0.0084	0.2501
X3	2	0.0261	0.7166	0.7171	0.0449	1.1309	1.1318
(E-W)	5	0.0065	0.4406	0.4406	0.0105	0.6872	0.6873
	8	0.0112	0.2423	0.2426	0.0191	0.3879	0.3884
	11	0.0040	0.1300	0.1301	0.0073	0.2500	0.2501
X4	2	8.533×10^{-5}	1.1517×10^{-4}	1.433×10^{-4}	1.485×10^{-4}	2.019×10^{-4}	2.506×10^{-4}
(Vert.)	5	6.523×10^{-5}	8.4598×10^{-5}	1.068×10^{-4}	1.076×10^{-4}	1.433×10^{-4}	1.742×10^{-4}
-Torsional	8	4.756×10^{-5}	6.3888×10^{-5}	7.965×10^{-5}	8.072×10^{-5}	1.091×10^{-4}	1.357×10^{-4}
	11	2.222×10^{-5}	2.6195×10^{-5}	3.435×10^{-5}	3.851×10^{-5}	4.711×10^{-5}	6.085×10^{-5}

b. Peak Nodal Accelerations Due To Vertical SSE AND OBE Excitations			
Vertical Nodal Acceleration (g)			
Elevation (Feet)	**Node No.**	**SSE**	**OBE**
108.00	1	0.39	0.21
	2	0.84	0.54
	3	0.33	0.18
	4	0.90	0.58
81.00	5	0.60	0.38
	6	0.86	0.51
	7	2.75	1.48
65.75	9	0.90	0.59
	10	0.24	0.14
	11	0.82	0.53
53.00	12	0.70	0.45
	13	1.61	1.01
	14	1.51	0.85
	15	0.19	0.09
	16	0.80	0.51
25.00	17	1.06	0.69
	18	0.69	0.43
	19	0.81	0.46
7.00	20	0.70	0.45
	21	0.53	0.31

TABLE 3.7(B)-14 PRIMARY AUXILIARY BUILDING SEISMIC ANALYSIS - PEAK HORIZONTAL

a. Peak Horizontal Responses - (RSS SUM)

Earthquake Direction	Beam No.	Shear Forces (Kips)		Bending Moments (Kip-in)		Torque (Kip-in)
		N-S Component	E-W Component	About N-S Axis	About E-W Axis	
	2	1463.1	57.2	18538	474040	22189
OBE	5	5532.2	56.4	31488	2315400	815930
	8	8371.0	127.0	46502	5107700	615340
X2	11	9258.9	161.5	74077	7092500	1133500
(N-S)						
	2	2319.5	98.3	31842	751520	37025
SSE	5	8609.8	91.4	53301	3606700	1273600
	8	13015.0	208.8	72437	7936600	979160
	11	14421.0	269.7	117370	11019000	1784200
	2	55.5	1569.1	508830	17980	27163
OBE	5	71.5	5633.5	2382200	38119	954730
	8	127.5	8079.5	5070200	59945	991020
X3	11	161.5	8594.0	6909800	86661	583090
(E-W)						
	2	96.0	2476.2	8023000	31097	47611
SSE	5	114.8	8764.2	3707900	63325	1492200
	8	209.0	12572.0	7878400	43963	1578500
	11	269.7	13409.0	10739000	136910	982020

b. Maximum Axial Forces Due to Vertical SSE and OBE Excitations		
	Axial Forces (Kips)	
Beam No.	**Safe Shutdown Earthquake**	**Operating Basis Earthquake**
1	671.46	385.70
2	2990.20	1644.40
3	2988.70	1643.70
4	4997.10	2747.50
5	5908.60	3251.30

TABLE 3.7(B)-15 CONTROL AND DIESEL GENERATOR BUILDING SEISMIC ANALYSIS - NATURAL FREQUENCIES, PERIODS AND MODE CLASSIFICATION

Mode Number	Natural Frequency (Hz)	Natural Period (second)	Mode Classification
1	5.6911	0.1757	Bending about X3 axis, (1st)
2	9.0942	0.1100	Bending about X2 axis, (1st)
3	11.6257	0.0860	Rotation about X1, (1st)
4	17.5683	0.0569	Bending about X3, (2nd)
5	18.4860	0.0541	Vertical, (1st)
6	23.8661	0.0419	Bending about X2, (2nd)
7	27.0055	0.0370	Bending about X3, (3rd)
8	31.1205	0.0321	Rotation about X1, (2nd)
9	33.2826	0.0300	Bending about X2, (3rd)
10	35.6255	0.0281	Vertical, (2nd)
11	39.6037	0.0253	Rotation about X1, (3rd)
12	53.2650	0.0188	Higher mode
13	67.5424	0.0148	Higher mode
14	68.5735	0.0146	Higher mode
15	80.4142	0.0124	Higher mode

TABLE 3.7(B)-16 **CONTROL AND DIESEL GENERATOR BUILDING SEISMIC ANALYSIS - PEAK NODAL**

a. Peak Nodal Acceleration

			SSE 7% Damping		**OBE 4% Damping**	
Earthquake Direction	**Node No.**	**Elevation (ft)**	**Spectrum Method (g)**	**Time Hist Method (g)**	**Spectrum Method (g)**	**Time Hist. Method (g)**
	3	98.0	0.482	0.416	0.270	0.234
X1	6	78.5	0.322	0.365	0.179	0.198
VERTICAL	9	75.0	0.392	0.388	0.219	0.215
	12	51.5	0.197	0.318	0.109	0.169
	15	50.0	0.233	0.330	0.130	0.178
X2	3	98.0	0.868	0.942	0.556	0.568
HORIZONTAL	6	78.5	0.513	0.606	0.328	0.353
(N-S)	9	75.0	0.588	0.688	0.377	0.403
	12	51.5	0.256	0.399	0.159	0.219
	15	50.0	0.319	0.454	0.197	0.259
X3	3	98.0	0.731	0.695	0.468	0.430
HORIZONTAL	6	78.5	0.599	0.607	0.385	0.358
(E-W)	9	75.0	0.581	0.594	0.373	0.349
	12	51.5	0.357	0.440	0.226	0.240
	15	50.0	0.343	0.433	0.217	0.234

a. Peak Nodal Acceleration

			SSE 7% Damping		OBE 4% Damping	
Earthquake Direction	**Node No.**	**Elevation (ft)**	**Spectrum Method (g)**	**Time Hist Method (g)**	**Spectrum Method (g)**	**Time Hist. Method (g)**
	3	98.0	0.01407	0.01373	0.00794	0.00765
X1	6	78.5	0.00948	0.00978	0.00542	0.00539
VERTICAL	9	75.0	0.01163	0.01152	0.00659	0.00640
	12	51.5	0.00587	0.00619	0.00339	0.00340
	15	50.0	0.00689	0.00703	0.00392	0.00389
X2	3	98.0	0.25722	0.30061	0.16554	0.17799
HORIZONTAL	6	78.5	0.14751	0.17781	0.09491	0.10448
(N-S)	9	75.0	0.17413	0.20552	0.11207	0.12135
	12	51.5	0.06150	0.07703	0.03954	0.04483
	15	50.0	0.07846	0.09474	0.05047	0.05561
X3	3	98.0	0.08531	0.08681	0.05480	0.05258
HORIZONTAL	6	78.5	0.07077	0.07296	0.04547	0.04384
(E-W)	9	75.0	0.06853	0.07076	0.04403	0.04249
	12	51.5	0.04042	0.04290	0.02596	0.02549
	15	50.0	0.03872	0.04113	0.02487	0.02443

TABLE 3.7(B)-17 CONTROL AND DIESEL GENERATOR BUILDING SEISMIC ANALYSIS - PEAK FORCES

a. Peak Forces Due to Horizontal OBE Excitation

Beam No.	Shear Forces (Kips)		Bending Moments (Kip-in)		Torque (Kip-in)
	N-S Component	E-W Component	About N-S Axis	About E-W Axis	
1	1758.7	1349.2	442940	1418000	245510
2	1861.9	2439.7	1005900	1793000	285010
3	2207.1	1824.4	287120	552170	81225
4	3934.7	3132.9	739260	1863900	335790
5	4274.2	3660.3	1200900	2245800	444710
6	2201.3	1878.8	268760	571200	95508
7	2229.4	2677.1	189430	207150	704940
8	3667.9	4573.9	905730	696150	417720
9	4536.6	5017.5	420560	1053300	158910
10	5502.6	6089.0	1613100	3977000	522740

b. Peak Forces Due to Horizontal SSE Excitation

Beam No.	Shear Forces (Kips)		Bending Moments (Kip-in)		Torque (Kip-in)
	N-S Component	E-W Component	About N-S Axis	About E-W Axis	
1	2739.4	2100.9	700770	2205500	382510
2	2908.7	3799.0	1571500	2786600	446310
3	3441.2	2851.6	448830	861370	129190
4	6115.6	4878.1	1157100	2897700	524710
5	6651.5	5699.8	1874100	3489600	695840
6	3439.1	2938.8	420350	891580	152550
7	3475.2	4170.5	333860	323520	1095900
8	5700.6	7122.1	1415300	1082000	652050
9	7051.7	7810.8	662930	1640000	260050
10	8561.9	9481.7	2514100	6179500	817920

c. Peak Axial Forces Due to Vertical SSE and OBE Excitations		
	Axial Forces (Kips)	
Beam No.	**Safe Shutdown Earthquake**	**Operating Basis Earthquake**
1	1558.0	901.3
2	2872.7	1732.2
3	1908.2	1076.5
4	3723.1	2127.0
5	5169.2	3045.1
6	1981.3	1109.8
7	2346.1	1314.6
8	4075.4	2282.9
9	4799.0	2696.7
10	5745.1	3221.3

TABLE 3.7(B)-18 FUEL STORAGE BUILDING SEISMIC ANALYSIS - NATURAL FREQUENCIES

a. Natural Frequencies, Periods and Mode Classification

Mode Number	**Natural Frequency (Hz)**	**Mode Classification**
1	10.35	1st Horizontal (N-S)
2	17.16	Mixed Mode
3	32.41	2nd Horizontal (N-S)
4	40.98	3rd Horizontal (N-S)
5	47.95	-
6	65.88	-
1	8.69	-
2	11.24	1st Horizontal (E-W)
3	18.07	Mixed Mode
4	38.64	2nd Horizontal (E-W)
5	52.10	3rd Horizontal (E-W)
6	65.62	-

b. Summary of Single Degree of Freedom Systems				
	(Vertical)			
	Floor Area: Roof, Filter Floor, MCC and RR Track Floor			
Elevation (feet)	Frequency (Hz)	Weight (Kips)	Location (Column Lines)	Element Type
84.0	4.0	850	C to D	slab/beam
	5.5	1203	whole area bet. A to C, except @ 2 and 3	slab/beam/ girder
	13.5	271	@ 2 and 3	
	4.0	563	whole area except @ 2 and 3	slab/beam/ girder
64.0	8.5	140	@ 2 and 3	beam
21.5, 20.5	9.0	178	B to C 3 to 4	slab/beam
	12.0	84	B to C 3 to 4	slab/beam
	10.0	430	B to C/3 to 4 A to B/2 to 3	slab/beam slab/beam
	20.0	90	A to B 2 to 3	slab/beam slab/beam

c. Natural Frequencies and Mode Classification - Crane Girder System		
Mode Number	**Natural Frequency (Hz)**	**Mode Classification**
1	10.49	1st Vertical Mode
2	18.16	2nd Vertical Mode
3	35.20	3rd Vertical Mode
4	40.34	4th Vertical Mode
5	50.34	-
6	55.63	-
7	85.93	-

TABLE 3.7(B)-19 FUEL STORAGE BUILDING SEISMIC ANALYSIS - PEAK NODAL

a. Peak Nodal Acceleration Due to Horizontal Excitations - (N-S Motion)					
			Translational Acceleration (g)		Rotational Acceleration About Vertical Axis (*)
			N-S Component		
Earthquake Motion	Mass Elevation (feet)	Node No.	Spectrum Method	Time History Method	
	84	2	.616	.641	9.10
		4	.484	.541	12.00
	64				
		5	.439	.479	8.08
SSE		6	.265	.345	9.67
	46.5				
		7	.303	.336	5.17
		8	.250	.286	5.51
	34.5				
		9	.250	.287	5.65
	25	12	.250	.253	1.73
	84	2	.385	.408	5.15
		4	.303	.332	6.80
	64				
		5	.268	.285	4.37
		6	.159	.200	5.13
OBE	46.5				
		7	.172	.192	2.78
		8	.125	.155	2.88
	34.5				
		9	.125	.154	2.88
	25	12	.125	.125	8.81

* (10^{-5} rad/sec^2)/(386.4 in/sec^2)

b. Peak Nodal Acceleration Due to Horizontal Excitations (E-W Motion)					
			Translational Acceleration (g)		**Rotational Acceleration About Vertical Axis (*)**
Earthquake Motion	**Mass Elevation (feet)**	**Node No.**	**N-S Component**		
			Spectrum Method	**Time History Method**	
	84	2	.471	.494	10.84
		5	.340	.382	8.85
	64				
		6	1.333	1.148	-
	46.5	8	.250	.301	7.97
SSE					
	34.5	9	.250	.270	6.38
	25	12	.250	.253	5.79
	84	2	.291	.286	6.33
		5	.207	.215	5.11
	64	6	.852	.710	-
	46.5	8	.127	.160	4.22
OBE					
	34.5	9	.125	.739	3.26
	25	12	.125	.128	2.90

* (10^{-5} rad/sec^2)/(386.4 in/sec^2)

c. Peak Nodal Accelerations Due to Vertical SSE and OBE Excitations					
		Vertical Nodal Acceleration (g)			
		Safe Shutdown Earthquake		**Operating Basis Earthquake**	
Mass Elevation (feet)	**Node No.**	**Spectrum Method**	**Time History Method**	**Spectrum Method**	**Time History Method**
	1	.726	.758	.468	.471
84	2	.624	.846	.402	.584
	3	.534	.506	.317	.296
	4	.301	.297	.151	.154
	5	.515	.753	.332	.466
64	6	.607	.589	.391	.397
	7	.248	.287	.124	.147
46.5	8	.171	.273	.085	.140
34.5	9	.167	.261	.003	.132
25.0	10	.167	.251	.083	.127
	13	.429	.461	.254	.268
21.5					
	14	.343	.335	.129	.186
	11	.567	.606	.365	.416
20.5	12	.473	.469	.288	.271
	13	.429	.461	.254	.260

d. Peak Nodal Accelerations Due to Vertical SSE and OBE Excitations (Crane Girder Support System)			
		Vertical Nodal Acceleration (g)	
Mass Elevation (feet)	**Node No.**	**SSE Time History Method**	**OBE Time History Method**
	2	.268	.136
	3	.311	.162
	4	.348	.184
	5	.452	.263
	6	.528	.329
64.0	7	.586	.372
	8	.569	.357
	9	.592	.377
	10	.538	.333
	11	.515	.313
	12	.399	.216
	13	.284	.145

e. Peak Nodal Displacements Due to Horizontal SSE and OBE Excitations

			Horizontal Nodal Displacements (inches) N-S Component	
Earthquake Motion	**Elevation (feet)**	**Mass Node No.**	**Spectrum Method**	**Time History Method**
	84	2	.0563	.0615
		4	.0440	.0486
	64			
		5	.0372	.0413
		6	.0215	.0242
SSE	46.5			
		7	.0195	.0221
		8	.0095	.0108
	34.5	9	.0087	.0100
	25	12	.0012	.0014
	84	2	.0351	.0382
		4	.0275	.0300
	64	5	.0233	.0255
		6	.0135	.0149
OBE	46.5	7	.0122	.0135
		8	.0059	.0066
	34.5	9	.0054	.0061
	25	12	.0008	.0009

f. Peak Nodal Displacements Due to Horizontal SSE and OBE Excitations				
			Horizontal Nodal Displacements (inches) E-W Component	
Earthquake Motion	**Elevation (feet)**	**Mass Node No.**	**Spectrum Method**	**Time History Method**
SSE	84	2	.0413	.0462
		5	.0284	.0322
	64	6	.1628	.1517
	46.5	8	.0145	.0169
	34.5	9	.0064	.0077
	25	12	.0009	.0012
OBE	84	2	.0258	.0271
		5	.0178	.0187
	64	6	.1045	.0960
	46.5	8	.0091	.0099
	34.5	9	.0040	.0045
	25	12	.0006	.0007

g. Peak Nodal Displacements Due to Vertical SSE and OBE Excitations					
		Vertical Nodal Displacements (inches)			
		Safe Shutdown Earthquake		Operating Basis Earthquake	
Mass Elevation (feet)	**Node No.**	**Spectrum Method**	**Time History Method**	**Spectrum Method**	**Time History Method**
84	4	.0031	.0047	.0018	.0026
64	7	.0025	.0037	.0014	.0021
46.5	8	.0015	.0023	.0008	.0013
34.5	9	.0007	.0012	.0004	.0007
25	10	.0001	.00019	.00006	.0001

TABLE 3.7(B)-20 FUEL STORAGE BUILDING SEISMIC ANALYSIS - MAXIMUM FORCES

a. Maximum Forces Due to Horizontal Excitations

Earthquake Motion	Beam No.		Shear Forces (Kips)		Bending Moments (Kip-in)		Torque (10^5 Kip-in)	
	N-S	E-W	N-S	E-W	About N-S Axis	About E-W Axis	N-S	E-W
SSE	1	1	1140	3126	750320	326920	2.438	4.377
	2	6	1977	3126	750320	419470	1.958	-
	3	2	2683	4685	1730400	889300	4.578	8.095
	4	7	2295	4685	1730400	901070	1.998	-
	5	3	2904	5144	2463600	1307100	4.706	7.386
	6	7	2546	5144	2463600	1266000	2.073	-
	7	4	2972	5297	3070500	1645600	4.752	7.516
	8	7	2637	5297	3070500	1565000	2.106	-
	9	5	5689	5334	3357200	3513800	6.851	5.278
OBE	1	1	712	1949	467650	204540	1.498	2.709
	2	6	1236	1949	467650	262560	1.225	-
	3	2	1680	2926	1080600	556990	2.866	5.03
	4	7	1437	2926	1080600	564240	1.249	-
	5	3	1819	3213	154220	818770	2.945	4.573
	6	7	1594	3213	154220	793030	1.294	-
	7	4	1862	3307	1918300	1030800	2.973	4.648
	8	7	1649	3307	1918300	980290	1.314	-
	9	5	3524	3327	2097500	2201100	4.286	3.218

b. Maximum Axial Forces Due to Vertical SSE and OBE Excitations		
	Axial Forces (Kips)	
Beam No.	**Safe Shutdown Earthquake**	**Operating Basis Earthquake**
1	1276	755
2	1985	1106
3	2245	1225
4	2345	1271
5	2367	1282

TABLE 3.7(B)-21 FREQUENCY INTERVAL FOR CALCULATING FLOOR RESPONSE SPECTRA

FREQUENCY RANGE, HZ.	INCREMENT
0.5 - 1.6	0.1
1.6 - 2.8	0.2
2.8 - 4.0	0.3
4.0 - 9.0	0.5
9.0 - 16.0	1.0
16.0 - 22.0	2.0
22.0 - 37.0	3.0
37.0 - 45.0	4.0

TABLE 3.7(B)-22 NON-CATEGORY I STRUCTURES DESIGNED AGAINST COLLAPSE ONTO ADJACENT CATEGORY I STRUCTURES DUE TO SSE LOADS

Non-Category I Structure	Affected Category I Structure
Turbine Building[(1)]	Condensate Storage Tank (E.W.) Containment and Others (N.S.)
Nonessential Switchgear Building[(2)]	Control & Diesel Generator Building
Tank Farm Area (Steel Framing Portion which Includes Steel Framing, Concrete Roofing and Metal Siding over Refueling Water Storage Tank)[(3)]	Primary Auxiliary Building, Waste Processing Building and Tank Farm Area Tunnels
Steam Generator Blowdown Recovery Building[(4)]	Primary Auxiliary Building, Waste Processing Building and Tank Farm
Circulating Water Pumphouse Steel Framing Portion[(5)]	Service Water Pumphouse
Waste Processing Building Steel Framing Portion and Reinforced Concrete Portion, Except the Area Between Columns 1 to 2 and A to D Between Elevations 53'-0" and 86'-0"[(6)]	Primary Auxiliary Building Tank Farm Area and Piping Tunnels

NOTES:

(1) The entire Turbine Building is designed against failure in the north-south direction. The south end is designed against failure in the east-west direction; and east-west failure in the north end will not affect any seismic Category I structures.

(2) The Nonessential Switchgear Building is designed mechanistically to fall away from the Control and Diesel Generator Building under the action of a collapsing Administration and Service Building. Thus, no significant load is applied to the Control and Diesel Generator Building by either the falling Administration and Service Building or the falling Nonessential Switchgear Building.

(3) The steel framing portion is designed and constructed such that the operating basis earthquake (OBE) and safe shutdown earthquake (SSE) would not cause the steel framing portion to collapse upon any safety-related structures, systems or components within or surrounding the Tank Farm area.

(4) The Steam Generator Blowdown Recovery Building is designed not to collapse.

(5) The collapse of the Circulating Water Pumphouse on the Service Water Pumphouse was evaluated to prove that the collapse will not impair the Service Water Pumphouse or system.

(6) The Waste Processing Building is designed not to collapse.

TABLE 3.7(B)-23 CRITICAL DAMPING VALUES

Item, Equipment or Component	Damping Percent Critica	
	OBE	**SSE**
Piping Systems	1	2
Valves, Compact Pumps, Compressors, Diesel Generators, Pipe Mounted Instrumentation	1	2
Heat Exchangers, Tanks & Vessels, Control Cabinets, Deep Well Pumps, Fans, Electrical Switchgear, Filters, Dampers, Motors	2	3
Electrical Conduits with Bolted Connections	4	7
Electrical Conduit with WeldedConnections	2	4
Equipment Supported bySeismic Dampers	8	8
Cable Trays (Refer to Subsection 3.7(B).1.3)		
NOTE: Higher damping values may be used provided that adequate justification is available		

TABLE 3.7(B)-24 SEISMIC INSTRUMENTATION LOCATION(S)

Item	Seismic Instrumentation	Location	Location Figure
1	Triaxial Time History Accelerograph, XT-6700	Free-field position in control room intake, on bedrock	3.7(B)-32
2	Triaxial Time History Accelerograph, XT-6701	Containment building foundation between Columns 16 and 17	1.2-2
3	Triaxial Time History Accelerograph, XT-6710	Containment Building between Columns 16 and 17 on the concrete operating floor	1.2-4
4	Triaxial Time History Accelerograph, XR-5707	Primary Auxiliary Building, elevation 53'-0"	1.2-9
5	Triaxial Time History Accelerograph, XR-6708	Service water pumphouse electrical room, west wall, north of column 12, at elevation 22'-0"	1.2-46

TABLE 3.7(N)-1 DAMPING VALUES USED BY WESTINGHOUSE FOR SEISMIC SYSTEMS ANALYSIS

	Damping (Percent of Critical)	
Item	**Upset Conditions (OBE)**	**Faulted Condition (SSE, DBA)**
Primary coolant loop system components and large piping[a]	2*	4*
Small piping	1*	2*
Welded steel structures	2	4
Bolted and/or riveted steel structures	4	7

[a]Applicable to 12 inch or larger diameter piping.

*The damping values of ASME B&PV Code, Code Case N-411, may be used for pipe stress verification and for pipe support optimization in place of the values given in Regulatory Guide 1.61.

TABLE 3.8-1 CONTAINMENT LOAD COMBINATIONS AND LOAD FACTORS(5)

Design Conditions	Category	Load Combination Number	Dead Load	Live Load	ASR Load	Test Pressure	Accident Pressure	Test Temperature	Normal Temperature	DBA Temperature	Operating Basis Earthquake	Safe Shutdown Earthquake	Wind Load	Tornado	Normal Pipe Reaction	DBA Thermal Pipe Reaction	R_r (DBA Local Effects): Reaction of Ruptured High Energy Pipe	R_r (DBA Local Effects): Jet Impingement Loads	R_r (DBA Local Effects): Impact of Ruptured High Energy Pipe	Pressure Variations	Design Basis Flood
	Loading Notation		D	L	S_a	P_t	P_a	T_t	T_o	T_a[1]	E_o	E_{ss}	W	W_t	R_o	R_a	R_{rr}	R_{rj}	R_{rm}	P_v[2]	F[3]
Service Load	Test	1	1.0	1.0	1.0	1.0	-	1.0	-	-	-	-	-	-	-	-	-	-	-	-	-
Service Load	Normal	2	1.0	1.0	1.0	-	-	-	1.0	-	-	-	-	-	1.0	-	-	-	-	1.0	-
Service Load	Severe Environmental	3	1.0	1.0	1.0	-	-	-	1.0	-	1.0	-	-	-	1.0	-	-	-	-	1.0	-
Factored Load	Severe Environmental	4	1.0	1.3	1.0	-	-	-	1.0	-	1.5	-	-	-	1.0	-	-	-	-	1.0	-
Factored Load	Extreme Environmental	5a	1.0	1.0	1.0	-	-	-	1.0	-	-	-	-	1.0[7]	1.0	-	-	-	-	1.0	1.0
		5b	1.0	1.0	1.0	-	-	-	1.0	-	-	1.0	-	1.0[7,8]	1.0	-	-	-	-	1.0	-
Factored Load	Abnormal	6a	1.0	1.0	1.0	-	1.5[4]	-	-	1.0[4]	-	-	-	-	-	1.0	-	-	-	-	-
		6b	1.0	1.0	1.0	-	1.0[4]	-	-	1.0[4]	-	-	-	-	-	1.25	-	-	-	-	-
Factored Load	Abnormal/Severe Environmental	7	1.0	1.0	1.0	-	1.25[4]	-	-	1.0[4]	1.25	-	-	-	-	1.0	1.0	1.0	1.0	-	-
Factored Load	Abnormal/Extreme Environmental	8	1.0	1.0	1.0	-	1.0[4]	-	-	1.0[4]	-	1.0	-	-		1.0	1.0	1.0	1.0	-	-

LOADING[6]

(1) Includes effect of normal operating thermal loads and accident loads. For all abnormal load conditions, structure should be checked to assure that accident pressure load without thermal load can be resisted by the structure within the specified allowable stresses for this condition.

(2) Negative pressure variations inside the structure shall not be considered simultaneously with outside negative pressure due to tornado loadings.

(3) For this load case, the design basis flood elevation shall be the max. ground water elevation, i.e., El. +20'-0".

(4) Load cases examined for maximum pressure and its coincident liner temperature and maximum liner temperature with its coincident pressure.

(5) All load factors shall be taken as 1.0 for the design of steel liner.

(6) See Subsection 3.8.1.3 for discussion of loadings.

(7) W_t includes missile effects only.

(8) For this load case, loadings from E_{ss} or W_t included individually.

TABLE 3.8-2 **ALLOWABLE STRESSES AND STRAINS IN THE CONTAINMENT STRUCTURE**

		SERVICE LOADS		**FACTORED LOADS**
		Mechanical Loads	**Thermal or Test Loads Added**	**All Loads**
fc' = 4000psi fc' = 3000psi	Concrete Compressive Stresses			
	Membrane	0.9 ksi	1.35 ksi	2.25 ksi
	Memb. + Bending	1.35 ksi	1.8 ksi	2.55 ksi
	Membrane	1.2 ksi	1.8 ksi	3.0 ksi
	Memb. + Bending	1.8 ksi	2.4 ksi	3.4 ksi
	Shear Stresses			
	Radial	CC-3431.3 & Code CASE N-287 (2)	CC-3431.3 & Code CASE N-287 (2)	CC-3431.4.1 & Code Case N-287 (2)
	Tangential	CC-3431.3	CC-3431.3	CC-3421.5.1
	Punching	CC3431.3 & Code Case N-219 (3)	CC3431.3 & Code Case N-219 (3)	CC-3421.6 & Code Case N-219 (3)
	Reinforcing (fy = 60 ksi) Tensile stress	30.0 ksi	45.0 ksi 40.0 ksi (4)	54.0 ksi (5)
	Liner (6)			
	Tensile Strain			
	Membrane	.002 in/in	.002 in/in	.003 in/in
	Memb. + Bending	.004 in/in	.004 in/in	.010 in/in
	Compressive Strain			
	Membrane	.002 in/in	.002 in/in	.005 in/in
	Memb. + Bending	.004 in/in	.004 in/in	.014 in/in
Liner Anchorage System Force/displacement limits are based on tests which are described in Appendix 3G and are in accordance with Table CC-3730-1				

NOTES:

(1) The allowable shear stress, vc, to be resisted by the concrete does not exceed 40 psi and 60 psi for load combinations 7 and 8, respectively, in Table 3.8-1.

(2) Radial shear reinforcement was designed per Code Case N-287.

(3) Peripheral shear reinforcement was designed per Code Case N-219.

(4) The allowable stresses for reinforcing steel are per Subarticle CC-3432.1, 1979 Summer Addendum.

(5) Stresses and strains in reinforcing steel are per Subarticle 3422.1, 1977 Winter Addendum.

(6) The liner allowables are per Table CC-3720-1, 1976 Winter Addendum.

TABLE 3.8-3 PROPERTIES OF MATERIALS FOR DESIGN OF CONTAINMENT

Concrete - 3000 psi

28-day Compressive Strength = 3000 psi

Modulus of Elasticity = 3.12×10^{6} psi

Coefficient of Thermal Expansion = 6.5×10^{-6} in./in./°F

Poissons' Ratio = 0.15

<u>Concrete - 4000 psi</u>

28-day Compressive Strength = 4000 psi

Modulus of Elasticity = 3.61×10^{6} psi

Coefficient of Thermal Expansion = 6.5×10^{-6} in./in./°F

Poissons' Ratio = 0.15

<u>Reinforcing Steel</u>

Design Yield Strength = 60,000 psi

Modulus of Elasticity = 29×10^{6} psi

Coefficient of Thermal Expansion = 6.5×10^{-6} in./in./°F

<u>Containment Liner</u>

Design Yield Strength = 32,000 psi

Modulus of Elasticity = 29×10^{6} psi

Coefficient of Thermal Expansion = 6.5×10^{-6} in./in./°F

Poissons' Ratio = 0.3

TABLE 3.8-4 CONTAINMENT LOAD CASES, BOUNDARY CONDITIONS, CRACK PATTERNS

Description of Load Case	Cracking Pattern	Bound Cond.	Liner Incl. in Stress Comp.	Liner Incl. in Force & Moment Comp.	Ref. Temp.
PA = 52 psi	Wall & Dome-Cracked; Mat - Uncracked	Base Z-Fixed Base R-Free	No	No	--
Dead Load	Wall & Dome-Cracked; Mat - Uncracked	Base Z-Fixed Base R-Free	No	No	--
To (Operating Temp. Gradient) (120°F, inside to (-) 10°F outside)	Dome-Wall Mat Inner Half - Uncracked Outer Half - Cracked	Base Z-Fixed Base R-Free	Yes	No	70°F
Hot Liner (Accident Temp. Spike) ΔT = 151.1°F	Wall & Dome - Cracked Mat - Uncracked	Base Z-Fixed Base R-Free	Yes	No	70°F
OBE	Shell Model - Uncracked	Bot. Of Wall - Fixed	--	--	--
SSE	Shell Model - Uncracked	Bot. Of Wall - Fixed	--	--	--

TABLE 3.8-5 BOUNDARY CONDITIONS FOR EQUIPMENT HATCH ANALYSIS

LOAD CASE	X-Z PLANE	Y-Z PLANE	X-Z PLANE	Y-Z PLANE
Dead Load	Sym.	Sym.	$v=\theta_2=0$	$u=\theta_1=0$
Pressure	Sym.	Sym.	$v=\theta_2=0$	$u=\theta_1=0$
Vert. (OBE)	Sym.	Sym.	$v=\theta_2=0$	$u=\theta_1=0$
Vert. (SSE)	Sym.	Sym.	$v=\theta_2=0$	$u=\theta_1=0$
N-S (OBE)	Sym.	Anti-Sym.	$v=\theta_2=0$	$v=w=\theta_2=0$
N-S (OBE)	Sym.	Anti-Sym.	$V=\theta_2=0$	$v=w=\theta_2=0$
E=W (OBE)	Anti-Sym.	Sym.	$u=w=\theta_1=0$	$v=\theta_1=0$
E-W (SSE)	Anti-Sym.	Sym.	$u=w=\theta_1=0$	$U=\theta_1=0$
NOTE: Base is fixed i.e., $u=v=w=\theta_1=\theta_2=0$				

TABLE 3.8-6 EQUIPMENT HATCH AND PERSONNEL LOCKS - LOAD COMBINATIONS AND LOAD FACTORS

		LOADING									
Category	Load Combination No.	Dead Load	Live Load	Test Pressure	Accident Pressure	Test Temperature	Operational Temperature	Accident Temperature	Operating Basis Earthquake	Safe Shutdown Earthquake	Pressure Variations
Loading Notation		**D**	**L**	P_t	P_a	T_t	T_o	T_a	**E**	**E'**	P_e
Test	1	1.0	1.0	1.0	---	1.0	---	---	---	---	---
Normal	2	1.0	1.0	---	---	---	1.0	---	---	---	1.0
Upset (Severe Environmental)	3	1.0	1.0	---	---	---	1.0	---	1.0	---	---
Faulted (Abnormal/ Severe Environmental)	4	1.0	1.0	---	1.0	---	---	1.0	1.0	---	---
Faulted(Abnormal/ Extreme Environmental)	5	1.0	1.0	---	1.0	---	---	1.0	---	1.0	---

NOTES:

1. Number of load cycles for test condition shall be as per 10 CFR 50 Appendix J.
2. The fatigue analysis is based on the consideration of 20 occurrences of Operating Basis Earthquake (OBE) each having 20 cycles of maximum response per seismic event.
3. The design of equipment hatch and personnel air locks are based on Tables 3.8-6 and 3.8-10 which are in agreement with the load combinations and stress limits defined in SRP 3.8.2, Revision 0, 11/14/75. It has been shown that the design of equipment hatch and personnel air locks also meets the current (Rev. 1, 7/81) SRP requirements.

TABLE 3.8-7 LOAD COMBINATIONS FOR DESIGN LOAD SET AND FOR ACTUAL APPLIED PIPE LOADS FOR MODERATE ENERGY PIPING PENETRATIONS

Design Condition	Load Combinations For Actual Pipe Loads Applied to Penetration
Design/Norma	P + D + T
Upset	P + D + T + OBE + SAD(OBE) + TH
Emergency/Faulted	P + D + T + SSE + SAD(SSE) + TH + PAD + TAD
P	Maximum Containment Pressure
D	Dead Weight
T	Thermal Load
OBE	Operating Basis Earthquake
SAD(OBE)	Seismic Anchor Displacement due to OBE
SSE	Safe Shutdown Earthquake
SAD(SSE)	Seismic Anchor Displacement due to SSE
TH	Thrust
PAD + TAD	-Pressure Plus Thermal Anchor Displacement

TABLE 3.8-8 ELECTRICAL PENETRATION ASSEMBLY LOAD COMBINATIONS AND LOAD FACTORS

CATEGORY	Load Combination No.	Dead Load	Test Pressure	Accident Pressure	Test Temperature	Operational Temperature	Accident Temperature	Operating Basis Earthquake	Safe Shutdown Earthquake	External Pressure
LOADING NOTATION		**D**	**P_t**	**P_a**	**T_t**	**T_o**	**T_a**	**E**	**E'**	**P_e**
TEST	a	1.0	1.0	--	1.0	--	--	--	--	--
NORMAL	b	1.0	--	--	--	1.0	--	--	--	1.0
UPSET (Severe Environmental)	c	1.0	--	--	--	1.0	--	1.0	--	--
FAULTED (Abnormal/Severe Environmental)	d	1.0	--	1.0	--	--	1.0	1.0	--	--
FAULTED (Abnormal/Extreme Environmental)	e	1.0	--	1.0	--	--	1.0	--	1.0	--

NOTES:
1. Number of load cycles for test condition shall be as per 10 CFR 50 Appendix J.
2. The fatigue analysis is based on the consideration of 20 occurrences of operating basis earthquake (OBE), each having 20 cycles of maximum response per seismic event.

TABLE 3.8-9 FUEL TRANSFER TUBE ASSEMBLY LOAD COMBINATIONS AND LOAD FACTORS

CATEGORY	Load Combination No.	Dead Load	Test Pressure	Operating Pressure	Accident Pressure	Test Temperature	Operational/Refueling Temperature	Accident Temperature	Operating Basis Earthquake	Safe Shutdown Earthquake	Live Load	Bellow Loads	Hydrostatic Pressure
LOADING NOTATION		**D**	P_t	P_o	P_a	T_t	T_o	T_a	E_o	E_s	**L**	**R**	**H**
TEST	1	1.0	1.0	--	--	1.0	--	--	--	--	--	1.0	--
NORMAL	2	1.0	--	1.0	--	--	1.0	--	--	--	--	--	--
NORMAL (Refueling)	3	1.0	--	--	--	--	1.0	--	--	--	1.0	--	1.0
UPSET (Severe Environmental)	4	1.0	--	--	--	--	1.0	--	1.0	--	--	1.0	--
FAULTED (Abnormal/Extreme Environmental)	5	1.0	--	--	1.0	--	--	1.0	--	1.0	--	1.0	--

NOTES:

1. The number of load cycles for test condition shall be as per 10 CFR 50 Appendix J.
2. The fatigue analysis is based on the consideration of 20 occurrences of the Operating Basis Earthquake, each occurrence having 20 cycles of maximum response.

TABLE 3.8-10 **STRESS LIMITS FOR EQUIPMENT HATCH AND PERSONNEL LOCKS**

		Primary Stresses					
	Load Combination No	**General Membrane P_m**	**Local Membrane P_L**	**Bending + Loca Membrane $P_B + P_L$**	**Primary & Secondary Stresses**	**Peak Stresses**	**Buckling**
TEST	1	0.9 S_y	1.25 S_y	1.25 S_y	3 S_m	Consider for Fatigue Analysis	125% of Allow. Given by NE-3133
CONSTRUCTION	2	S_m	1.5 S_m	1.5 S_m	3 S_m	Consider for Fatigue Analysis	Allow. Given By NE-3133
NORMAL	3	S_m	1.5 S_m	1.5 S_m	3 S_m	Consider for Fatigue Analysis	Allow. Given By NE-3133
UPSET (SEVERE ENVIRONMENTAL)	4	S_m	1.5 S_m	1.5 S_m	3 S_m	Consider for Fatigue Analysis	Allow. Given By NE-3133
FAULTED (ABNORMAL/ SEVERE ENVIRONMENTAL)	5	S_m	1.5 S_m	1.5 S_m	N/A	N/A	Allow. Given By NE-3133
FAULTED (ABNORMAL/ EXTREME ENVIRONMENTAL	6a*	S_m	1.5 S_m	1.5 S_m	N/A	N/A	Allow. Given By NE-3133
	6b**	The Greater of 1.2 S_m or S_y	The Greater of 1.8 S_m or 1.5 S_y	The Greater of 1.8 S_m or 1.5 S_y	N/A	N/A	120% Allow. Given NE-3133

		Primary Stresses					
	Load Combination No	**General Membrane P_m**	**Local Membrane P_L**	**Bending + Loca Membrane $P_B + P_L$**	**Primary & Secondary Stresses**	**Peak Stresses**	**Buckling**

NOTES: (1) Thermal stresses need not be considered in computing P_m, P_L and P_B.

(2) Thermal effects are considered in:

(a) Specifying stress intensity limits as a function of temperature.

(b) Analyzing effects of cyclic operation.

(3) * - Not Integral and Continuous

** - Integral and Continuous

(4) S_y = yield stress

S_m = allowable primary membrane stress

TABLE 3.8-11 PROPERTIES OF MATERIALS FOR PERSONNEL AIR LOCKS AND EQUIPMENT HATCH

1. Plates

a. Personnel Air Lock - Pressure retaining areas and welded pipe

SA-516, Gr. 70 Properties @ 271°F:
$E = 27.4 \times 10^6$ psi
$S_y = 33,961$ psi
$S_m = 19,300$ psi
$S_{br} = 33,961$ psi
$S_s = 15,440$ psi

SA-283-C Properties @ 120°F:
$S_y = 30,000$ psi
$S_m = 12,600$ psi
$S_{br} = S_y = 30,000$ psi
$S_s = 0.8\ S_m = 10,800$ psi

b. Personnel Air Lock - Tapered wedges for door locking

SA-240, Type 304 Properties @ 271°F:
$S_y = 30,000$ psi (110°F)
$S_y = 23,225$ psi
$S_{br} = S_y = 23,225$ psi
$S_m = 18,700$ psi

c. Equipment Hatch - Pressure retaining areas and welded pipe

SA-516, Gr. 60 Properties @ 271°F:	SA-516, Gr. 60 Properties @ 120°F:
$E = 27.49 \times 10^6$	$E = 27.82 \times 10^6$ psi
$S_y = 28,560$ psi	$S_y = 31,440$ psi
$S_m = 16,500$ psi	$S_m = 16,500$ psi
$\alpha = 6.54 \times 10^{-6}$ in/in/°F	$\alpha = 6.18 \times 10^{-6}$ in/in/°F

2. Bolts

SA-193, Gr. B7 Properties @ 271°F:
$S_y = 105,000$ psi (110°F)
$S_y = 94,100$ psi
$S_m = 27,500$ psi
$S_s = 22,000$ psi
$S_{br} = 94,100$ psi

3. Pins

C1045 Properties @ 120°F:
$S_y = 59,000$ psi
$S_m = 0.6S_y = 35,400$ psi
$S_s = 0.4S_y = 23,600$ psi
$S_{br} = 0.9S_y = 53,100$ psi

(18-8) Type 304 (PAL* in Equipment Hatch) Properties @ 271°F:
$S_y = 30,000$ psi (110°F)
$S_y = 22,500$ psi
$S_m = 18,300$ psi
$S_{br} = S_y = 22,500$ psi

4. Pipes

SA-333, Gr. 6 Properties @ 271°F:

S_y = 31,261 psi
S_m = 15,000 psi
S_s = 12,000 psi
S_{br} = S_y = 31,261 psi

5. Seals

Silicone or approved equal

6. Welding Filler Materials

Welding Process	ASME Filler Metal Specification
SMAW	SFA-5.1 Class E7018
	SFA-5.1 Class E6010
	SFA-5.4 Class E309
SAW	SFA-5.17 Electrode Class-ELX-ELXX
	-EMXX-EMXXX
	-EHXX
	Flux Class-F6X-xxx
	-F7X-xxx
GMAW-GTAW	SFA-5.18 Electrode Class-E705-X
	SFA-5.9 Electrode Class-FR-309

*PAL - Personnel Air Lock

TABLE 3.8-12 ALLOWABLE STRESS INTENSITIES FOR HIGH ENERGY PIPING PENETRATIONS

Condition	Loading	Allowable Stress Intensity[1]
Design (including normal, upset and emergency conditions)[2]	Pipe operating pressure + mechanical loads (including seismic)	$P_m \leq Sh$
	Pipe operating pressure + mechanical loads (including seismic)	$P_1 + P_b \leq 1.5\ Sh$
	Pipe operating pressure and temperature + mechanical loads (including seismic)	$P_1 + P_b + Q \leq 3\ Sh$
Faulted	Pipe operating pressure + mechanical loads (including seismic)	$P_m \leq$ min. of 2.4 Sh or 0.7 Su
	Pipe operating pressure + mechanical loads(including seismic)	$P_1 + P_b \leq$ min. of 3.6 Sh or 1.05 Su

[1] The symbols P_m, P_1, P_b, Sh and Su are defined in the ASME B&PV Code, Section III, Division 1, Subsection NA.

[2] Concrete temperatures are checked for compliance with the limits of the ASME B&PV Code, Section III, Division 2, Subsection CC.

TABLE 3.8-13 **MATERIALS FOR PIPING PENETRATIONS, ELECTRICAL PENETRATIONS, FUEL TRANSFER TUBE ASSEMBLY AND VENTILATION PENETRATION ASSEMBLIES**

1. HIGH ENERGY PIPING PENETRATIONS		
a.	Process Pipe and Flued Head	
		SA105
	or	SA182-304
	or	SA182-316
b.	Sleeve	
		SA333, Gr.1
2. MODERATE ENERGY PIPING PENETRATIONS		
a.	Process Pipe	
		SA312-304
	or	SA106, Gr.B
	or	SA376-316
b.	End Plate	
		SA516, Gr.60
c.	Sleeve	
		SA333, Gr.1
3. ELECTRICAL PENETRATIONS		
a.	Installation Weld	
		SA333, Gr.6
b.	Monitoring Plate (Bulkhead) and Clamps	
		SA240-304
c.	Module	
		SA479-304
d.	Studs and Bolts	
		SA193-B7
e.	Other Materials	
	An epoxy sealant and "O" ring seal are also used in the electrical penetrations.	

4. FUEL TRANSFER TUBE ASSEMBLY

a. Bolting Flange

SA182-304

b. Bolts

SA164, Gr.400, Cond. A, H.F.

c. Tube

SA240-304

d. Quick Closure Hatch, Mounting Ring and Slip-on Flange

SA182-304

e. Other Materials

Two self-energizing elastomer quad rings are also used in the fuel transfer tube assembly.

5. VENTILATION PENETRATION ASSEMBLIES

a. Sleeve

SA333, Gr. 1

b. Flange

SA105

c. CAP Penetration Blind Flanges

SA516, Gr. 70

d. Studs and Nuts

SA 193, Grade B7

SA 194, Grade 7, 2H, 3, or 4

e. Other Materials

A resilient seal material (EPDM) is used for the blind flange o-rings.

TABLE 3.8-14 INTERIOR CONTAINMENT STRUCTURES BASIC LOAD COMBINATIONS AND LOAD FACTORS

Design Conditions	Material / Loading Notations		Load Case Number	Dead Load and Hydrostatic Load	Live Load	ASR Loads	Accident Pressure	Operational Temperature	Accident Temperature	Operating Basis Earthquake	Safe Shutdown Earthquake	Operational Piping Loads	Accident Piping Loads	Jet Force Reaction	Jet Impingement Loads	Missile Impact Loads	Internal Missile Loads	Stress Limit or Design Criteria
				LOADING[1]														
				D	L	S_a	P_a	T_o	T_a	E_o	E_{ss}	R_o	R_a	R_{rr}	R_{rj}	R_{rm}	M	
Normal Load	Structural Steel		1S	1.0	1.0	-	-	-	-	-	-	-	-	-	-	-	-	$\leq F_s$ Per AISC
			2S	1.0	1.0	-	-	-	-	1.0	-	-	-	-	-	-	-	
			3S	0.67	0.67	-	-	0.67	-	-	-	0.67	-	-	-	-	-	
			4S	0.67	0.67	-	-	0.67	-	0.67	-	0.67	-	-	-	-	-	
	Concrete		1C	1.4	1.7	2.0	-	-	-	-	-	-	-	-	-	-	-	ACI 318-71
			2C	1.4	1.7	1.3	-	-	-	1.9	-	-	-	-	-	-	-	
			3C	-	-	-	-	-	-	-	-	-	-	-	-	-	-	
			4C	1.05	1.28	1.0	-	1.28	-	1.43	-	1.28	-	-	-	-	-	
Unusual Load	Structural Steel [3] [4]	Elastic	5S	0.63	0.63	-	-	0.63	-	-	0.63	0.63	-	-	-	-	-	$\leq F_s$ Per AISC
			6S	0.63	0.63	-	0.63	-	0.63	-	-	-	0.63	-	-	-	-	
			7S	0.63	0.63	-	0.63	-	0.63	0.63	-	-	0.63	0.63	0.63	0.63	0.63	
			8S	0.59	0.59	-	0.59	-	0.59	-	0.59	-	0.59	0.59	0.59	0.59	0.59	
	[3] [4]	Plastic	5S	1.1	1.1	-	-	1.1	-	-	1.1	1.1	-	-	-	-	-	AISC, Part II
			6S	1.1	1.1	-	1.7	-	1.1	-	-	-	1.1	-	-	-	-	
			7S	1.1	1.1	-	1.4	-	1.1	1.4	-	-	1.1	1.1	1.1	1.1	1.1	
			8S	1.1	1.1	-	1.1	-	1.1	-	1.1	-	1.1	1.1	1.1	1.1	1.1	
	Concrete [4]		5C	1.0	1.0	1.0	-	1.0	-	-	1.0	1.0	-	-	-	-	-	ACI 318-71
			6C	1.0	1.0	1.0	1.5	-	1.0	-	-	-	1.0	-	-	-	-	
			7C	1.0	1.0	1.0	1.25	-	1.0	1.25	-	-	1.0	1.0	1.0	1.0	1.1	
			8C	1.0	1.0	1.0	1.0	-	1.0	-	1.0	-	1.0	1.0	1.0	1.0	1.0	

(F_s = Allowable Stress)

(1) See Subsection 3.8.3.3 for discussion of loadings.

(2) In above load combinations, the peak values of P_a, T_a, R_a, R_{rj}, R_{rr}, R_{rm} and M shall be combined (when they act concurrently) unless time history analysis is performed to justify otherwise.

(3) For these load combinations either elastic or plastic design may be used.

(4) Load combinations 7S, 8S, 7C and 8C are also checked without R_{rr}, R_{rj}, R_{rm}.

TABLE 3.8-15 DESCRIPTIONS OF SEISMIC CATEGORY I STRUCTURES

	Foundation				Building				
Structure	Length	Width	Thickness	Comments	Length	Width	Height	Comments	Figures
Containment Enclosure Building	-	10'-3"	10'	Ring with Inside Diameter = 153'-0" & Outside Diameter = 173'-6"	-	162'-6"	230' A		1.2-2 through1.2-6
Containment Equipment Hatch Missile Shield									1.2-4, 1.2-5
Containment Structure	-	153' diam	10'	Also includes pit for reactor	-	140' I.D.	219' A		1.2-2 through 1.2-6
Containment Internal Structures	See Containment Structure				See Subsection 3.8.3				1.2-2 through 1.2-6
Containment Enclosure Ventilation Area	116'-0"	55'-9" Maximum	2'-6"	Overall Dimensions	116'-0"	55'-9" Maximum	31'-6"	Overall dimensions (Pipe tunnel below not included)	1.2-2 through1.2-4
Control and Diesel Generator Building	138'-0" A 95'-0" A	90'-0" 92'-0"	4', 3'-6" & 5'-5"	Stepped fdn. CB Stepped fdn. DGB	138'-0" 95'-0"	90'-0" 90'-0"	119'-3" A 95"-0" A		1.2-31 through 1.2-23 CB 1.2-34 through1.2-36 DGB
Control Room Makeup Air Intake Structure	20'-4"	20'-4"	3'-0"	-	18'-4"	18'-4"	10'-9"		3.8-31
Emergency Feedwater Pump Building including Electric Cable Tunnels and Penetration Areas (Control Building to Containment)	86'	44'-6¼" maximum	4'	Overall Dimensions	84'		41'-3¼"	73'	1.2-2 through
Enclosure for Condensate Storage Tank		29'-0" diam.	5'-0"			21'-3 I.D.		Shield wall around tank	3.8-32
Fuel Storage Building	100' A	98'-6"	2' to 4'-4¾"	Stepped fdn.	98'		97'-6"	110' A	1.2-15 through 1.2-21

	Foundation				Building				
Structure	**Length**	**Width**	**Thickness**	**Comments**	**Length**	**Width**	**Height**	**Comments**	**Figures**
Main Steam and Feedwater Pipe Chase (East) including East Penetration Area	133'-7"	30'-3" maximum	6'-0"		127'-7"	22'-3" maximum	61'-6"		1.2-3, 1.2-4
Main Steam and Feedwater Pipe Chase (West) including Mechanical Penetration Area and Personnel Hatch Area	127'-9" 45'-9"	34'-8½" 37'-0" maximum	6'-0" 13'-0"	Chase Hatch Area (Fill Conc)	123'-9" 45'-9"	20'-0" 37'-0"	99'-0" maximum 29'-3"	Chase and Mechanical Penetration Area Hatch Area (irregular shape)	1.2-2 through 1.2-5
Piping Tunnels	192'15' approx. total	2'-3" min.		Thickness varies	192' approx. total	14'	10'-11" min.	Height varies	3.8-33
Pre-Action Valve Building	47'-10"	35'-0"	2'-2" to 4'-0"	Stepped fdn. overall dimensions	34'-10"	34'-6"	27'-4"	Overall dimensions	1.2-51
Primary Auxiliary Building including Residual Heat Removal Equipment Vault	147' A 62'-6"	81' A 48'-6"	4' 4'	Stepped fdn. Actually part of PAB foundation	145' 57'-0"	79' 43'-0"	101' 86'-6"	PAB RHR Equip. Vault	1.2-9 through 1.2-12 1.2-13, 1.2-14
Safety-Related Electrical Duct Banks and Manholes	22'-6" to 24'-0"	19'-0" to 23'-6"	2'-6"	Manholes-overall dimensions-several sizes used for Duct banks	18'-6" to 20'-0"	15'-0" to 19'-6"	9'-11" to 20'-8"	Manholes-overall dimensions-several sizes used for Duct banks	3.8-34 3.8-35

	Foundation				Building				
Structure	**Length**	**Width**	**Thickness**	**Comments**	**Length**	**Width**	**Height**	**Comments**	**Figures**
Service Water Access Vault	39'-0"	15'-0"	1'-6"		38'-0"	14'-0"	14'-0"		3.8-37
Service Water Cooling Tower including Switchgear Room	311'-9"	61'	4'		307'	56'	85'-6"		1.2-56
Service and Circulating Water Pumphouse C	74'-6" B 39'-4" 122'-0"	91'-2" 54'-0" 110'-8"	6' 3' 6'	SW PH Stepped fdn. SW PH Elec. Equip. CW PH	78'-0" 37'-4" 122'-0"	118'-8" 52'-0" 118'-8"	92'-0" 19'-6" 92'-0"	SW PH SW PH Elec. Equip. Rm CW PH	1.2-46 through 1.2-48
Tank Farm. Area (Tunnels) including Dikes and Foundations for Refueling Water Storage Tank and Reactor Makeup Water Storage Tank	156' A	65'-10"	 4'-5" 20'	Overall dimensions Under pipe chase Under Tanks	152' 63' 67'	65'-10"	92' 58'-4" 50'-10"	Overall dimensions 10' 22'	1.2-23 through 1.2-25, 1.2-27 MUW Tank Dike RWS Tank Dike
Waste Processing Building	188'-9" 29'-9'	86'-3" 23'-6"	4' 28'	Stepped fdn. fill concrete	188'-9" 29'-9"	86'-3" 23'-6"	89' to 142', 61'	Main structure Decontamination room (In main structure, foundation and roof are irregular)	1.2-22 through 1.2-30

A - Approximate dimension

B - Outer wall is thickened to provide full building width (78'-0") at grade.

C - Circulating Water Pumphouse is attached to flumes and intake and discharge transition structures.

TABLE 3.8-16 **CATEGORY I, STRUCTURES OTHER THAN REACTOR CONTAINMENT STRUCTURE OR ITS INTERNALS BASIC LOAD COMBINATIONS AND LOAD FACTORS**

Design Conditions	Material		Required Strength	Loading(1), (4): All Structures: Dead Load and Hydrostatic Load	All Structures: Live Load	All Structures: Operational Temperature	All Structures: Operating Basis Earthquake	All Structures: Safe Shutdown Earthquake	All Structures: Wind	All Structures: Tornado Wind	Concrete Only: Lateral Earth Pressures	Concrete Only: ASR Loads (6)	Concrete Only: Earth Pressure due to OBE	Concrete Only: Earth Pressure due to SSE	Certain Structures, Where Appropriate: Operational Piping Loads	Accident Pressure	Accident Piping Loads	Pipe Break Loads (Rr): Jet Impingement Loads	Pipe Break Loads (Rr): Missile Impact Loads	Pipe Break Loads (Rr): Jet Force Reaction	Internal Missile Loads	Accident Temperature	Design basis Flood	Unusual Snow Load	Stress Limit (5) Or Design Criteria
	Loading Notations			D	L	T_o	E_o	E_s	W	W_t	E	S_a	He	H_s	R_o	P_a	R_a	R_{rj}	R_{rm}	R_{rr}	M	T_a	F	Ls	
Normal Load	Structural Steel		S	1.0	1.0	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-			$\leq F_s$ Per AISC
				1.0	1.0	-	1.0	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-			
				1.0	1.0	-	-	-	1.0	-	-	-	-	-	-	-	-	-	-	-	-	-			
				0.67	0.67	0.67	-	-	-	-	-	-	-	-	0.67	-	-	-	-	-	-	-			
				0.67	0.67	0.67	0.67	-	-	-	-	-	-	-	0.67	-	-	-	-	-	-	-			
				0.67	0.67	0.67	-	-	0.67	-	-	-	-	-	0.67	-	-	-	-	-	-	-			
	Concrete		U	1.4	1.7	-	-	-	-	-	1.7	2.0	-	-	-	-	-	-	-	-	-	-			ACI 318-71
				1.4	1.7	-	1.9	-	-	-	1.7	1.3	1.9	-	-	-	-	-	-	-	-	-			
				1.4	1.7	-	-	-	1.7	-	1.7	1.7	-	-	-	-	-	-	-	-	-	-			
				1.05	1.28	1.28	-	-	-	-	1.28	1.5	-	-	1.28	-	-	-	-	-	-	-			
				1.05	1.28	1.28	1.43	-	-	-	1.28	1.0	1.43	-	1.28	-	-	-	-	-	-	-			
				1.05	1.28	1.28	-	-	1.3	-	1.28	1.28	-	-	1.28	-	-	-	-	-	-	-			
				1.2	-	-	-	-	1.7	-	1.7	1.7	-		-	-	-	-	-	-	-	-			
				1.2	-	-	1.9	-	-	-	1.7	1.3	1.9		-	-	-	-	-	-	-	-			
Unusual Load	Structural Steel	Elastic		0.63	0.63	0.63	-	0.63	-	-	-	-	-	-	0.63	-	-	-	-	-	-	-			$\leq F_s$ Per AISC
			S	0.63	0.63	0.63	-	-	-	0.63	-	-	-	-	0.63	-	-	-	-	-	-	-			
			(2)	0.63	0.63	-	-	-	-	-	-	-	-	-	-	0.63	0.63	-	-	-	-	0.63			
			(2)	0.63	0.63	-	0.63	-	-	-	-	-	-	-	-	0.63	0.63	0.63	0.63	0.63	0.63	0.63			
				0.59	0.59	-	-	0.59	-	-	-	-	-	-	-	0.59	0.59	0.59	0.29	0.29	0.29	0.59			
		Plastic		1.1	1.1	1.1	-	1.1	--	-	-	-	-	-	1.1	-	-	-	-	-	-	-			AISC, Part II
			Y	1.1	1.1	1.1	-	-	-	1.1	-	-	-	-	1.1	-	-	-	-	-	-	-			
				1.1	1.1	-	-	-	-	-	-	-	-	-	-	1.7	1.1	1.1	1.1	1.1	1.1	1.1			
			(*)	1.1	1.1	-	1.4	-	-	-	-	-	-	-	-	1.4	1.1	-	-	-	-	1.1			
			(*)	1.1	1.1	-	-	1.1	-	-	-	-	-	-	-	1.1	1.1	1.1	1.1	1.1	1.1	1.1			
	Concrete		U	1.0	1.0	1.0	-	1.0	-	-	1.0	1.0	-	1.0	1.0	-	-	-	-	-	-	-			ACI 318-71
				1.0	1.0	1.0	-	-	-	1.0	1.0	1.0	-	-	1.0	-	-	-	-	-	-	-			
				1.0	1.0	-	-	-	-	-	1.0	1.0	-	-	-	1.5	1.0	-	-	-	1.1	1.0			
				1.0	1.0	-	1.25	-	-	-	1.0	1.0	1.25	-	-	1.25	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	1.0	1.0	1.0			
				1.0	1.0	1.0	-	-	1.0	-	-	1.0	-	-	1.0	-	-	-	-	-	-	-	(3)	1.0	
				1.0	1.0	1.0	-	-	1.0	-	-	1.0	-	-	1.0	-	-	-	-	-	-	-	1.0	1.0	

(1) In above load combinations, the peak values of P_a, T_a, R_a, R_{rj}, R_{rr}, R_{rm} and M shall be combined (when acting concurrently) unless time history analysis is performed to justify otherwise.
(2) Elastic cases to be checked for overall stability by the plastic load combination cases as indicated by (*).
(3) For design bases flood load case, elevation shall be the effective maximum ground water elevation i.e., El. + 20′-0″.
(4) See Subsection 3.8.4.3 for discussion of loadings.
(5) (F_s = Allowable Stress)
(6) Where ASR strains are greater than 0.05% (0.5 mm/m), ASR load factors may be reduced by 20% but not taken as less than 1.

TABLE 3.8-17 **COMPUTER PROGRAMS USED IN THE ANALYSIS AND DESIGN OF OTHER SEISMIC CATEGORY I STRUCTURES**

Computer Program	Structures On Which Used
1. MRI/STARDYNE (Static Analysis)	Control & Diesel Generator Building
	Fuel Storage Building
	Main Steam and Feedwater Pipe Chase (East)
	Main Steam and Feedwater Pipe Chase (West)
	Pre-Action Valve Area
	Primary Auxiliary Building Including Residual Heat Removal Equipment Vault
	Service Water Cooling Towers Including Switchgear Room
	Service Water Pumphouse
2. MARC-CDC (Static Analysis)	Containment Enclosure Building
3. LESCAL (Design of Reinforcing Steel)	Containment Enclosure Building
	Main Steam and Feedwater Pipe Chase (East)
4. GENSAP (Static Analysis)	Containment Enclosure Ventilation Area
	Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Areas
	Piping Tunnels
5. MULTISPAN (Static Analysis)	Service Water Cooling Towers
6. ANSYS (ASR Deformation Analysis)	All Seismic Category I Structures

TABLE 3.8-18 ASR EXPANSION LIMITS FOR STRUCTURAL LIMIT STATES

Structural Limit State	ASR Expansion Limit (4)
Shear	Through-thickness: See FP#101020 Section 2.1 (Ref 7)(2) Volumetric: See FP#101050 Appendix B (Ref 8) (3)
Flexure	
Reinforcement Anchorage	
Anchors	See FP #101020 Section 2.1 (Ref 7)
Compression	(1)

(1) Compressive load from ASR in the direction of reinforcement is combined and evaluated with other applied compressive loads.

(2) The through-thickness expansion limit for shear, flexure and reinforcement anchorage presented in FP #101020 (Reference 7) are different. The most limiting value is applied as the acceptance criterion for through-thickness expansion monitoring among these structural limit states.

(3) The maximum observed maximum volumetric expansion for shear, flexure and reinforcement anchorage identified in FP#101050 (Reference 8), Appendix B, Section 5 are different. The most limiting value is applied as the acceptance criterion for volumetric expansion monitoring among these structural limit states.

(4) NextEra Energy Seabrook will perform the following actions to confirm that expansion behavior at Seabrook Station aligns with observations from the MPR/FSEL test programs and that the associated expansion limits are applicable:

   a. Conduct assessments of expansion behavior to confirm that expansion behavior at Seabrook Station is comparable to what was observed in the MPR/FSEL test programs and to check margin for future expansion. Alignment confirms that the MPR/FSEL test program limits listed above are appropriate and applicable. NextEra Energy Seabrook completed the first expansion assessment in March 2018; subsequent expansion assessments will be performed every ten years, using the approach provided in FP#101050 (Reference 8) Appendix B.

   b. Corroborate the modulus-expansion correlation used to calculate pre-instrument through thickness expansion (FP#l00918; Reference 9)) for 20 percent of extensometer locations no later than 2025 and 10 years thereafter, using the approach provided in FP#101050 (Reference 8) Appendix C. The corroboration study involves obtaining new cores from the vicinity of 20 percent of the extensometers, determining the elastic modulus, and using the correlation to estimate total through-thickness expansion. This value will be compared to the expansion determined using the sum of the differential expansion measured by the extensometer and the pre-instrument expansion (using the correlation) at the time the extensometer was installed. Corroboration between the expansion values calculated using the modulus-expansion correlation with the expansion values determined from the extensometer and the original modulus-expansion result validates use of the correlation and confirms applicability of the MPR/FSEL test program limits.

(5) Through-thickness expansion is determined as the sum of expansion measured using installed instrumentation (e.g. extensometers) and the expansion that occurred prior to installation of the instrumentation ("pre-instrument" expansion). Pre-instrument through-thickness expansion is determined using the methodology defined in FP# 100918 (Reference 9).

TABLE 3. 9(B)-1 SYSTEMS REQUIRING PREOPERATIONAL VIBRATION TESTING

System	Flow Modes for Preoperational Vibration Testing	
	Steady State	Transient
Reactor Coolant	Single and multiple pump operation	Pump(s) starts and stops
		Pressurizer PORV discharge
Residual Heat Removal	Shutdown cooling mode	None
	Provide letdown when RCS pressure is low	
	Low head injection through RCS cold legs	
Safety Injection	Intermediate head injection	None
Chemical and Volume Control	Letdown flow modes	None
	Provide seal water injection	
	Normal purification	
	High head injection	
Primary Component Cooling	Loop A pump(s) operating	None
	Loop B pump(s) operating	
	Post LOCA recirculation	

Spent Fuel Pool Cooling	Pump(s) A and B operating	None
Service Water	Loop A pumps operating Loop B pumps operating	None

Flow Modes for Preoperational Vibration Testing

System	Steady State	Transient
Steam Generator Blowdown	Flow at normal rate Flow at maximum rate	Open/close containment isolation valve(s)
Condensate	Two pumps operating	None
Feedwater	Feedwater pump(s) operating FW pump recirculation	SGFP trip Containment isolation valve close
Emergency Feedwater	EFW pump operating	None
Main Steam	100% power operation	Turbine trip and dump and ARV Emergency feedwater pump trip (Terry turbine) Main steam headers to condenser dump lines
Containment Building Spray	Pump(s) A & B operating in minimum recirculation mode back to RWST	None

Diesel Generator	Fuel oil transfer pump(s) operating	None
	Starting air compressors operating	
	Cooling water system operating	
	Backup control air compressor operating	

TABLE 3.9B-2 **SYSTEMS REQUIRING MONITORING OF THERMAL EXPANSION DURING STARTUP FUNCTIONAL TESTING (NORMAL OPERATING TEMPERATURE FOR THE FOLLOWING SYSTEMS IS 200°F OR HIGHER)**

SYSTEM		CLASS	LOCATION/COMMENTS
Reactor Coolant	(RC)	1	All Piping Within the RCS Pressure Boundary
Reactor Heat	(RH)	1	All Piping Within the RCS Pressure Boundary
Safety Injection	(SI)	1	All Piping Within the RCS Pressure Boundary
Chemical & Volume Control	(CS)	1	All Piping Within the RCS Pressure Boundary
Reactor Coolant	(RC)	2	RCP Suction to REG. HX (CS-E-2) RC Hot Leg 1 to RH-P-8A Suction RC Hot Leg 4 to RH-P-8B Suction
Residual Heat	(RH)	2	RH-P-8A Discharge to RH-E-9A RH-P-8B Discharge to RH-E-9B RH-E-9A Outlet to RC Cold Leg 1 RH-E-9B Outlet to RC Cold Leg 4
Chemical & Volume Control	(CS)	2	REG. HX (CS-E-2) Outlet to LTDN HX (CS-E-4) REG. HX (CS-E-2) Outlet to RC Cold Loop 1&4 RCP Disch. to EX LTDN HX (CS-E-3)
Main Steam	(MS)	2	Stm. Gen. (RC-E-11A,B,C&D) to Class Break 2 N
Feedwater	(FW)	2	Stm. Gen. (RC-E-11A,B,C&D) to Class Break 2 N
Steam Blowdown	(SB)	2	Stm. Gen. (RC-E-11A,B,C&D) to SB-V9,10,11&12
Main Steam	(MS)	N	High Press. & Low Press. Steam to Moisture Separators & Turbines
Feedwater	(FW)	N	Feedwater Pumps (FW-P-32A,B)Discharge to Class Break 2 N Steam Generator Blowdown to Stm. Gen. & Wet Lay-Up Pump Stm. Gen. & Wet Lay-up Pump Disch. to Feedwater Pump (FW-P-32A,B) Discharge Side

SYSTEM		CLASS	LOCATION/COMMENTS
Condensate	(CO)	N	Heaters CO-E-22A, B&C to Heaters CO-E-23A, B&C: CO-E-24A,B&C; CO-E-25A&B Heaters CO-E-25A&B to Feedwater Pumps (FW-P-32A,B)
Steam Extraction	(EX)	N	High & Low Press. Extraction Steam to Heaters CO-E-23A,B&C; CO-E-24A,B&C; CO-E-25A&B; FW-E-26A&B
Heater Drains	(HD)	N	Heaters CO-E-24A,B&C; CO-E23A,B&C to Heaters CO-E-22A,B&C Heaters CO-E-25A&B; FW-E-26A&B to Heater Drain Tank (TK-22) Heater Drain Tank (HD-TK-22) to Heater Drain Pumps (HD-P31A&B) Suction Heater Drain Pump(HD-P31A&B) Discharge to Steam Generator Feed Pumps (FW-P-32A&B) Suction
Steam Blowdown	(SB)		SB-V9,10,11&12 to Flash Tank (SB-TK-40)

NOTE: Additional systems such as those listed below may be monitored or visual inspected for any constraints during startup & testing:

Main Steam Drains	(MSD)
Auxiliary Steam	(AS)
Auxiliary Steam Condensate	(ASC)
Moisture Separator Drains & Vents	(MD)
Heater Misc. Vents & Drains	(MVD)
Turbine Steam Seal System	(TSS)

TABLE 3.9(B)-3 DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 1, 2 AND 3 COMPONENTS AND SUPPORTS (OTHER THAN PIPING SYSTEMS)

Plant Loading Conditions	Design Loading Combinations
Normal	P + D + L + M + NOL + T
Upset	P + D + L + M + NOL + OBE + H + F_t + T
Emergency	P + D + L + M + NEL + H + F_t
Faulted	P + D + L + M + NFL + SSE + H + F_t

Where:

**P	=	Pressure corresponding to the loading condition
D	=	Dead weight
L	=	Live weight of fluid handled; for liquid content of vessel or tank
M	=	Snow or wind load for outdoor storage tank
NOL	=	Nozzle load for Normal/Upset condition
NEL	=	Nozzle load for Emergency condition
NFL	=	Nozzle load for Faulted condition
OBE	=	Operating Basis Earthquake (inertia load)
SSE	=	Safe Shutdown Earthquake (inertia load)
H	=	Dynamic fluid head effects (where applicable)
F_t	=	Valve thrust loads (where applicable)
*T	=	Thermal load (where applicable)

* Temperature for Class 2 and 3 is used to determine allowable stress only.

** Pressure for Class 1 includes LOCA effects.

TABLE 3.9(B)-4 STRESS LIMITS FOR NONACTIVE CATEGORY I, ASME CODE CLASS 2 AND 3 PUMPS

Plant Loading Conditions[1]		Stress Limits
Normal	ASME III, NC-3400 or ND-3400	
	P_m	1.1S
Upset	$(P_m \text{ or } P_1) + P_b$	1.65S
	P_m	1.5S
Emergency	$(P_m \text{ or } P_1) + P_b$	1.8S
	P_m	2.0S
Faulted	$(P_m \text{ or } P_1) + P_b$	2.4S

Where:

S = Material allowable stress at maximum temperature from Appendix I of ASME Section III

P_m = Primary general membrane stress, the average primary stress across the solid section under consideration. Excludes effects of discontinuities and concentrations. Produced by pressure and mechanical loads.

P_b = Primary bending stress. This stress is produced by pressure and mechanical loads including inertia earthquake effects but excluding effects of discontinuities and concentrations.

P_1 = Primary local membrane stress, the average stress across any solid section under consideration. Same as P_m except that discontinuities are considered.

NOTES

(1) Plant loading conditions are defined in Table 3.9(B)-3.

TABLE 3.9(B)-5 STRESS LIMITS FOR NONACTIVE CATEGORY I, ASME CODE CLASS 2 AND 3 VALVES

Plant Loading Conditions [1]	Stress Limits[2][3]
Normal	ASME III, NC-3500 or ND-3500
	$P_m < 1.1S$
Upset	$(P_m \text{ or } P_1) + P_b < 1.65S$
	$P_m < 1.5S$
Emergency	$(P_m \text{ or } P_1) + P_b < 1.8S$
	$P_m < 2.0S$
Faulted	$(P_m \text{ or } P_1) + P_b < 2.4S$

NOTES

(1) Plant loading conditions are defined in Table 3.9(B)-3.

(2) S, P_m, P_1 and defined in Table 3.9(B)-4.

(3) Reference Code Case 1635-1

TABLE 3.9(B)-6 STRESS LIMITS FOR NONACTIVE CATEGORY I ASME CODE CLASS 1 VALVES

Plant Loading Condition (1)	Stress Limits (2),(3),(4),(5),(6)
Normal	ASME III, Subsection NB-3500
	$P_m < 1.1\ S_m$
Upset	$(P_m \text{ or } P_1) + P_b < 1.65\ S_m$
	$S_n < 3\ S_m$
	$P_m < 1.5\ S_m$
Emergency	$(P_m \text{ or } P_1) + P_b < 1.8\ S_m$
	$P_m < 2.0\ S_m$
Faulted	$(P_m \text{ or } P_1) + P_b < 2.4\ S_m$

NOTES:

(1) Plant loading conditions are defined in Table 3.9(B)-3. Pressure loading includes LOCA effects.

(2) P_1, P_m and P_b are defined in Table 3.9(B)-4.

(3) S_m = Material stress intensity at maximum temperature form Appendix I of ASME III.

(4) Stress in the valve resulting form connecting pipe nozzle loads and internal pressure induced stresses should not exceed the limits listed in this table.

(5) Reference ASME Section III, Subsection NB-3221.

(6) Requirements of ASME III, Subsection NB-3540 must be met. Code Case 1552 is an acceptable alternate.

TABLE 3.9(B)-7 STRESS LIMITS FOR CATEGORY I ASME CODE CLASS 2 AND 3 VESSELS AND TANKS*

Plant Loading Conditions[(1)]	Stress Limits [(2)]
Normal	ASME NC-3300, ND-3300 NC-3800, ND-3800
Upset	$P_m < 1.1S$ $(P_m \text{ or } P_1) + P_b < 1.65S$
Emergency	$P_m < 1.5S$ $(P_m \text{ or } P_1) + P_b < 1.8S$
Faulted	$P_m < 2.0S$ $(P_m \text{ or } P_1) + P_b < 2.4S$

* Notes

(1) Plant loading conditions are defined in Table 3.9(B)-3.

(2) S, P_m, P_1 and P_b are defined in Table 3.9(B)-4.

TABLE 3.9(B)-8 ASME SECTION III CLASS 1 PIPING SYSTEMS LOAD COMBINATIONS AND STRESS LIMITS(1)

Condition	Load Combination	Stress Category	Stress Limits	Stress Combination
DESIGN	P	P_m	S_m	NB-3640
	P + D + OBE	$P_L + P_b$	1.5 S_m	EQ(9) NB-3650
PLANT NORMAL AND UPSET	P_{MAX}+T+TR+OBE+SAD(OBE)+ Q	$P_L+P_b+P_e+Q$	3 S_m	EQ(10) NB-3650
	OR			
	⎡T	P_e	3 S_m	EQ(12) NB-3650
	⎣P_{MAX} + TR + OBE + Q	P_L+P_b+Q	3 S_m	EQ(13) NB-3650
	AND			
	U	----	1.0	NB-3222.4
PLANT EMERGENCY	P_{MAX}	P_m	1.5xP(DESIGN)	NB-3655
	P_{MAX} + D + TR + DSL	$P_L + P_b$	2.25 S_m	EQ(9) NB-3655
PLANT FAULTED	P_{MAX}	P_m	2 x P(DESIGN)	NB-3656
	P_{MAX}+D+TR+SSE+SAD+(SSE)+ DSL+ LOCA DISP.	$P_L + P_b$	3 S_m	EQ(9) NB-3656
TEST	$P_t + D_t$	$P_m + P_b$	0.9 Sy	NB-3226
	PAD + D	P_b	1.35 Sy	NB-3226

(1) Terminology and notations are defined in Table 3.9(B)-11

TABLE 3.9B-9 ASME SECTION III CLASS 2 AND 3 ESSENTIAL PIPING SYSTEMS LOAD COMBINATIONS AND STRESS LIMITS(1)

Condition	Load Combination	Stress Combination	Stress Limits
DESIGN	P	NC-3640	S_h
PLANT NORMAL	⌈P + D ⌊T	EQ(8) NC-3650 EQ(10) NC-3650	S_h S_A
	OR		
	⌈P + D ⌊P + D + T	EQ(8) NC-3650 EQ(11) NC-3650	S_h $S_h + S_A$
PLANT UPSET	⌈P_{MAX}+D+TR+OBE+SAD(OBE) ⌊T	EQ(9) NC-3650 EQ(10) NC-3650	1.2 S_h S_A
	OR		
	⌈P_{MAX}+D+TR+OBE+SAD(OBE) ⌊P_{MAX}+ D + T	EQ(9) EQ(11)	1.2 S_h $S_h + S_A$
	OR		
	⌈P_{MAX} + D + TR + OBE ⌊T + SAD(OBE)	EQ(9) NC-3650 EQ(10) NC-3650	1.2 S_h S_A
	OR		
	⌈P_{MAX} + D + TR + OBE ⌊P_{MAX} + D + T + SAD(OBE)	EQ(9) NC-3650 EQ(11) NC-3650	1.2 S_h $S_h + S_A$
	AND		
	P_{MAX} + D	EQ(8) NC-3650	S_h

Condition	Load Combination	Stress Combination	Stress Limits
PLANT EMERGENCY	⎡P_{MAX} + D + TR + DSL ⎣T	EQ(9) NC-3650 EQ(11) NC-3650	1.2 S_h S_A
	OR		
	⎡P_{MAX} + D + TR + DSL ⎣P_{MAX} + D + T	EQ(9) NC-3650 EQ(11) NC-3650	1.2 S_h $S_h + S_A$
	AND		
	P_{MAX} + D	EQ(8) NC-3650	S_h
PLANT FAULTED	P_{MAX}	CC-1606	2 x P(DESIGN)
	⎡P_{MAX}+D+TR+SSE+SAD(SSE)+ PAD+TAD+DSL ⎣T	EQ(9) NC-3650 EQ(10) NC-3650	1.8 S_h S_A
	OR		
	⎡P_{MAX}+D+TR+SSE+SAD(SSE)+ ⎢ PAD+TAD+DSL ⎣P_{MAX} + D + T	EQ(9) NC-3650 EQ(11) NC-3650	1.8 S_h $S_h + S_A$
	OR		
	⎡P_{MAX}+D+TR+SSE+DSL ⎣T+SAD(SSE)+PAD+TAD	EQ(9) NC-3650 EQ(10) NC-3650	1.8 S_h S_A
	OR		

Condition	Load Combination	Stress Combination	Stress Limits
	⌈P_{MAX}+D+TR+SSE+DSL	EQ(9) NC-3650	1.8 S_h
	⌊P_{MAX}+D+T+SAD(SSE)+PAD+TAD	EQ(11) NC-3650	$S_h + S_A$
	AND		
	P_{MAX} + D	EQ(8) NC-3650	S_h
TEST	$P_t + D_t$	Adopted From NB-3226	0.9 Sy
	PAD + D		1.35 Sy

(1) Terminology and notations are defined in Table 3.9(B)-11.

TABLE 3.9(B)-10 ASME SECTION III CLASS 2 AND 3 NONESSENTIAL PIPING SYSTEMS[1]

Condition	Load Combination	Stress Combination	Stress Limits
DESIGN	P	NC-3640	S_h
PLANT NORMAL	⎡P + D ⎣T	EQ(8) NC-3650 EQ(10) NC-3650	S_h S_A
	OR		
	⎡P + D ⎣P + D + T	EQ(8) NC-3650 EQ(11) NC-3650	S_h $S_h + S_A$
PLANT UPSET	⎡P_{MAX}+D+TR+OBE+SAD(OBE) ⎣T	EQ(9) NC-3650 EQ(10) NC-3650	1.2 S_h S_A
	OR		
	⎡P_{MAX} + D + TR + OBE ⎣T + SAD(OBE)	EQ(9) NC-3650 EQ(10) NC-3650	1.2 S_h S_A
	OR		
	⎡P_{MAX} + D + TR + OBE ⎣P_{MAX} + D + T + SAD(OBE)	EQ(9) NC-3650 EQ(11) NC-3650	1.2 S_h $S_h + S_A$
	AND		
	P_{MAX} + D	EQ(8) NC-3650	S_h
PLANT EMERGENCY	P + D + TR + DSL	EQ(9) NC-3650	1.8 S_h
PLANT FAULTED	P + D + TR + SSE + DSL	EQ(9) NC-3650 CC-1606	2.4 S_h
TEST	$P_t + D_t$ PAD + D	Adopted Form NB-3226	0.9 Sy 1.35 Sy

(1) Terminology and notations are defined in Table 3.9(B)-11

TABLE 3.9B-11 TERMINOLOGY AND NOTATIONS USED IN TABLES 3.9(B)-8, 3.9(B)-9 AND 3.9(B)-10

Symbols for Stress Classification and Stress Limits are in accordance with ASME Section III. Other load symbols and definitions are specified below:

P - Internal design pressure

P_{MAX} - Peak pressure, considered as a set pressure of over-pressure safety devices

P_t - Test pressure

D - Deadweight, consisting of the weight of the pipe and pipe supported elements such as valves and flanges, including weight of insulation and contained fluid

D_t - Same as 'D' where pipe contents are fluid during pressure test

T - Thermal loads due to:

a. Piping thermal expansion when subjected to maximum temperature difference between the fluid and the surrounding environment in the specified plant conditions, and

b. Anchor displacement due to thermal movements of piping anchors

TR - Thrust or transient due to safety valve discharge, valve trip or fluid flow

SAD - Seismic anchor displacement (OBE or SSE), affects piping supported from different structures of relative seismic motions

PAD - Anchor displacement due to pressure, e.g., containment building penetrations due to internal pressure during test or LOCA

TAD - Anchor displacement due to thermal growth of the structure e.g., radial and vertical growth of Containment Building during LOCA ±MSL-

DSL - Dynamic System Load, Accident Load affecting piping as follows:

- Impact from missiles or pipe whip
- Jet impingement
- External pressure

LOCA - Anchor displacement due to movement of primary or secondary loop

DISP. during LOCA

Q - Temperature gradient loads, ΔT_1, ΔT_2 and $(\alpha_a T_a - \alpha_b T_b)$

U - Cumulative usage factor

OBE - Loads generated by the Operating Basis Earthquake (OBE), which is the earthquake that could reasonably be expected to affect the plant site during the operating life of the plant and which produces the vibratory ground motion for which those features of the nuclear plant necessary for continued operation without undue risk to the health and safety of the public have been designed to remain functional.

SSE - Loads generated by the Safe Shutdown Earthquake (SSE) which is the earthquake that produces the maximum vibratory ground motion for which certain structures, systems, and components important to safety and required for safe shutdown on the plant have been designed to remain functional.

TABLE 3.9(B)-12 STRESS LIMITS FOR ACTIVE CATEGORY I ASME CODE CLASS 2 AND 3 PUMPS

Plant Loading Condition [1][3]	Stress Limits [2][4]
Normal	ASME Section III, Subsections NC-3400 or ND-3400
Upset	P_m <1.0S
	$(P_m$ or $P_1) + P_b$ <1.5S
Emergency	P_m <1.0S
	$(P_m$ or $P_1) + P_b$ <1.5S
Faulted	P_m <1.0S
	$(P_m$ or $P_1) + P_b$ <1.5S

NOTES

(1) Plant loading conditions are defined in Table 3.9(B)-3.

(2) S, P_1, P_m and P_b are defined in Table 3.9(B)-4.

(3) Identification of the specific transients or events to be considered under each plant condition are addressed in Regulatory Guide 1.48.

(4) For pump supports, the allowable stresses defined in AISC "Manual of Steel Construction" is used for plant condition associated with 0.5SSE. For plant conditions associated with SSE, the stresses are limited to 90 percent of yield stress for the material involved.

TABLE 3.9(B)-13 STRESS LIMITS FOR ACTIVE CATEGORY I ASME CODE CLASS 2 AND 3 VALVES

Plant Loading Condition[1][3]	Stress Limits[2][3][4]
Normal	ASME Section III, Subsections NC-3500 or ND-3500
Upset	$P_m < 1.0S$
	$(P_m \text{ or } P_1) + P_b < 1.5S$
Emergency	$P_m < 1.0S$
	$(P_m \text{ or } P_1) + P_b < 1.5S$
Faulted	$P_m < 1.0S$
	$(P_m \text{ or } P_1) + P_b < 1.5S$

NOTES

(1) Plant loading conditions are defined in Table 3.9(B)-3.

(2) S, P_1, P_m and P_b are defined in Table 3.9(B)-4.

(3) Stress in the valve resulting from connecting pipe nozzle loads and internal pressure induced stresses should not exceed the limits listed in this table.

(4) Reference ASME Section III, Subsection NB-3221

TABLE 3.9(B)-14 STRESS LIMITS FOR ACTIVE CATEGORY I ASME CODE CLASS 1 VALVES

Plant Loading Condition [1]	Stress Limits [2],[3],[4],[5],[6]
Normal	ASME III, Subsection NB-3500
Upset	$P_m < 1.0$ Sm
	$(P_m \text{ or } P_1) + P_b < 1.5$ Sm
	$S_n < 3$ Sm
Emergency	$P_m < 1.0$ Sm
	$(P_m \text{ or } P_1) + P_b < 1.5$ Sm
Faulted	$P_m < 1.0$ Sm
	$(P_m \text{ or } P_1) + P_b < 1.5$ Sm
	$S_n < 3$ Sm

NOTES

(1) Plant loading conditions are defined in Table 3.9(B)-3. Pressure loading includes LOCA effects.

(2) P_1, P_m and P_b are defined in Table 3.9(B)-4.

(3) S_m = Material stress intensity at maximum temperature form Appendix I of ASME III.

(4) Stress in the valve resulting from connecting pipe nozzle loads and internal pressure induced stresses should not exceed the limits listed in this table.

(5) Reference ASME Section III, Subsection NB-3221.

(6) Requirements of ASME III, Subsection NB-3540 must be met. Code Case 1552 is an acceptable alternate.

TABLE 3.9(B)-15 STRESS AND DEFLECTION ANALYSIS OF PRIMARY COMPONENT COOLING WATER PUMPS (14X23-S)

ITEMS - SSE ANALYSIS	ACTUAL	ALLOWABLE
Pump Casing, Primary Membrane, psi	5,746	17,500
Membrane and Bending, psi	15,715	26,250
Casing Flange, Normal Stress, psi	24,469	26,250
Casing Flange Bolts, psi	34,967	50,000
Suction Flange, Longitudinal Stress, psi	10,213	21,000
Discharge Flange, Longitudinal Stress, psi	9,093	21,000
Shaft Stress, psi	6,724	25,000
Shaft Deflection, at Coupling, in.	0.017	0.055
at Seal, in.	0.004	0.005
at Impeller, in.	0.009	0.0125
Casing Feet, Principal, psi	3,535	14,000
Casing Foot Bolts, Tension, psi	22,280	42,000
Casing Foot Shear Pins, Shear, psi	14,328	17,000
Bedplate, Principal psi	13,990	19,333
Bedplate, Side-Channels, tension, psi	9,534	14,500
Bedplate, Side-Channel Weld, Shear, psi	9,154	9,670
Anchor Bolts, Tension, psi	12,090	19,100
Shear, psi	8,910	12,800
Natural Frequency of Pump-Motor-Bedplate System, Hz	46.19	N/A

TABLE 3.9(B)-16 STRESS AND DEFLECTION SUMMARY FOR CONTAINMENT SPRAY PUMPS (6X11X14B-CD)

Items - Sse Analysis	Actual	Allowable
Natural Frequency, Hz		
Pump	67	-
Driver	85	-
Pump Casing at Suction Nozzle, psi	2,741	27,000
Shaft Stress, psi	4,486	27,000
Pedestal Weld, psi	5,350	32,400
Pump Anchor Pin, psi	10,500	32,400
Base Cross Member, psi	20,374	32,400
Weld, psi	19,200	32,400
Bearing Load (Double Row), lbs.	1,140	17,200
(Single Row), lbs.	228	7,670
Base Hold-down bolts, Tension, psi	3,450	19,100
Shear, psi	2,430	9,900
Deflections		
Pump shaft, in.	0.0038	0.022
Coupling parallel misalignment, in.	0.0385	0.102
Coupling angular misalignment, degrees	0.264	1.5
Motor rotor, in.	0.0011	0.043

TABLE 3.9(B)-17 STRESS AND DEFLECTION SUMMARY FOR EMERGENCY FEEDWATER PUMPS (4x9 NH-10)

Items - See Analysis	Actual	Allowable
A. Turbine-Driven Pump		
Natural frequency of pump-turbine, Hz		
Entire assembly	48.0	-
Shaft	93.2	-
Deflection, in.		
Shaft @ seal	0.0042	0.005
Coupling	0.0081	0.102
B. Motor-Driven Pump		
Natural frequency of pump-motor		
Entire assembly, Hz	49.3	-
Shaft, Hz	93.2	-
Pump pedestal weld, psi		
Top	1,900	18,000
Bottom	15,100	18,000
Deflection, in.		
Shaft @ seal	0.0042	0.005
Coupling	0.0081	0.102

TABLE 3.9(B)-18 STRESS AND DEFLECTION SUMMARY FOR SERVICE WATER PUMP (INGERSOLL-DRESSER PUMP 42APK)

Items - See Analysis	Actual	Allowable
Natural Frequency, Hz		
Horizontal	9.7	-
Vertical	> 29	-
Pump Casing, psi		
Discharge Head at Nozzle	8,046	15,700
Column at Flange	10,460	28,500
Bowl	7,635	28,350
Lineshaft, psi	9,060	21,500
Seismic Support, psi	2,448	13,345
Base Bolts, psi	16,711	25,000
Clearance, in.		
Top Impeller Wear Ring	0.006	0.0145
Lower Impeller Wear Ring	0.006	0.0145
Motor Air Gap	0.001808	0.035

TABLE 3.9(B)-19 STRESS AND DEFLECTION SUMMARY FOR COOLING TOWER PUMP (INGERSOLL-DRESSER PUMP 29LKX)

Items - See Analysis	Actual	Allowable
Natural Frequency, Hz		
Pump Assembly	14.6	-
Motor Rotor	49.7	-
Pump Casing, psi		
Discharge Head Flange	19,437	20,550
Column Flange	21,960	28,500
Pump Casing	9,833	28,850
Shaft, psi	13,205	21,500
Seismic Supports, psi	891	23,550
Anchor Bolts, Tension, psi	16,416	17,800
Shear, psi	4,558	10,000
Clearances, in.		
Impeller to Casing	0.00	0.01
Rotor/Stator	0.017	0.03

TABLE 3.9(B)-20 STRESS AND DEFLECTION SUMMARY FOR DIESEL FUEL OIL TRANSFER PUMP (N 3DBS - 8 IMO PUMP)

ITEMS - SSE ANALYSIS	ACTUAL	ALLOWABLE
Natural Frequency, Hz	228	-
Pump Casing at Inlet Nozzle, psi	7,123	26,250
Base Plate, psi	4,147	21,750
Pump Hold-Down Bolts, psi		
Tension	2,300	19,100
Shear	1,150	9,900
Clearances, in.	0.00009	0.0005

TABLE 3.9(B)-21 STRESS ANALYSIS SUMMARY ASME IIICLASS 1 PIPING RCS PRESSURIZER SAFETY AND RELIEF VALVES SYSTEM

(P&ID RC-20846)

(LINE NOS: 74, 75, 76 & 80)

EVALUATION	MAX STRESS (KSI)	LINE NO. & COMPONENT	ALLOWABLE (KSI)
Eq. 9 - Design P+D+OBE	23.2	Line 76 Elbow	1.5 Sm 24.63
Eq. 9 - Faulted P+D+TR+SSE	39.1	Line 76 Elbow	3.0 Sm 49.26
Eq. 12 T	36.0	Line 80 Transition	3.0 Sm 49.26
Eq. 13 P+D+OBE+Q	37.9	Line 80 Transition	3.0 Sm 49.26
Fatigue Usage Factor	0.95	Line 80 Transition	1.0

*This table is being maintained for historical purposes only. Refer to applicable calculations for latest information.

TABLE 3.9B-22 STRESS ANALYSIS SUMMARY, ASME III CLASS 2 AND 3 PIPING (EQUATION 9)

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
* This table is being maintained for historical purposes only. Refer to applicable calculations for latest information.							
REACTOR COOLANT SYSTEM							
3"x4"- RC-V24	RC-20841	16.36	19.44	9.09	19.44	26.68	29.16
Line No. 14-1							
SC 2							
3"x4"- RC-V89	RC-20844	18.27	19.44	8.64	19.44	27.91	29.16
Line No. 88-1							
SC 2							
REFUELING CAVITY CLEANUP SYSTEM							
3/4"x1" - SF-V101	SF-20484	13.24	19.08	1.17	19.08	21.66	28.62
Line No. 1743-9							
SC 2							

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
FLOOR & EQUIPMENT DRAIN SYSTEM							
3/4"x1"- WLD-V209 **Line No. 2076-11** **SC 2**	WLD-20219	10.5	22.0	2.3	22.0	21.81	33.0
1"x3/4" - RH-V13 **Line No. 172-1** **SC 2**	RH-20662	11.41	19.44	1.97	19.44	20.87	29.16
1"x3/4" - RH-V25 **Line No. 169-1** **SC 2**	RH-20663	13.18	19.44	2.73	19.44	23.12	29.16
3/4"x1" - SI-V247 **Line No. 248-16** **SC 2**	SI-20450	13.79	19.08	4.40	19.08	23.19	28.62
1"x2" - SI-V10 **Line No. 220-1** **SC 2**	SI-20450	9.58	19.92	7.56	19.92	11.60	29.88

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
1"x2" - SI-V30 Line No. 221-1 SC 2	SI-20450	9.77	19.92	4.91	19.92	14.62	29.88
1"x2" - SI-V45 Line No. 222-1 SC 2	SI-20450	9.77	19.92	4.91	19.92	14.62	29.88
1"x2" - SI-V60 Line No. 223-1 SC 2	SI-20450	9.77	19.92	4.91	19.92	14.69	29.88
3/4"x1" - SI-V101 Line No. 252-1 SC 2	SI-20446	15.23	19.92	4.08	19.92	26.34	29.88
3/4"x1" - SI-V113 Line No. 253-1 SC 2	SI-20446	14.88	19.92	3.54	19.92	26.06	29.88
3/4"x1" - SI-V76 Line No. 254-1 SC 2	SI-20447	10.61	19.92	4.41	19.92	17.25	29.88

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
3/4"x1" - SI-V175	SI-20447	15.43	22.56	4.48	22.56	26.38	33.84
Line No. 278-2							
SC 2							
3/4"x1" - SI-V248	SI-20446	4.34	19.68	3.38	19.68	5.31	29.52
Line No. 255-10							
SC 2							
3/4"x1" - CBS-V62	SI-20446	8.74	19.92	2.02	19.92	15.46	29.88
Line No. 1228-1							
SC 2							
CHEMICAL & VOLUME CONTROL SYSTEM							
3"x2" - CS-V148	CS-20722	2.12	20.64	1.68	20.64	6.35	30.96
Line No. 353-1							
SC 2							
3/4"x1" - CS-V794	CS-20726	0.82	20.64	0.35	20.64	1.29	30.96
Line No. 332-26							
SC 2							

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
2"x3" - CS-V173 **Line No. 353-1** **SC 2**	CS-20726	2.12	20.64	1.68	20.64	2.55	30.96
4"x3" - CS-V243 **Line No. 384** **SC 2**	CS-20725	3.11	19.54	1.88	19.54	4.94	29.30
1"x3/4" - CS-V492 **Line No. 372-1** **SC 2**	CS-20725	17.59	22.06	6.03	22.06	24.53	33.09
1"x3/4" - CS-V227 **Line No. 373-1** **SC 2**	CS-20725	6.21	19.92	1.08	19.92	11.33	29.88
3"x2" - CS-V250 **Line N-378-1** **SC 2**	CS-20726	8.10	19.54	4.43	19.54	11.77	29.30

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
SPENT FUEL POOL COOLING AND CLEANUP SYSTEM							
1"x3/4" - SF-V74	SF-20482	3.14	19.92	1.58	19.92	4.7	29.88
Line No. 1741-1							
SC 3							
1"x3/4" - SF-V45	SF-20482	3.14	19.92	1.58	19.92	4.7	29.88
Line No. 1723-1							
SC 3							
1"x3/4" - SF-V183	SF-20482	4.04	21.19	2.56	21.19	6.75	31.78
Line No. 1766-1							
SC 3							

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
PRIMARY COMPONENT COOLING WATER SYSTEM							
3/4"x1" - CC-V320	CC-20211	0.97	18.00	0.35	18.00	1.60	27.00
Line No. 807-2							
SC 3							
3/4"x1" - CC-V321	CC-20211	8.45	18.00	0.35	18.00	16.55	27.00
Line No. 809-5							
SC 3							
3"x4" - CC-V486	CC-20207	14.47	18.00	2.11	18.00	22.94	27.00
Line No. 777-31							
SC 3							
1½"x2½" - CC-V326	CC-20209	6.15	18.00	4.92	18.00	7.37	27.00
Line No. 714-4							
SC 3							
1½"x2" - CC-V364	CC-20207	5.87	18.00	4.00	18.00	7.75	27.00
Line No. 713-7							
SC 3							

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
1½"x2½" - CC-V242 Line No. 718-4 SC 3	CC-20209	5.65	18.00	5.11	18.00	6.19	27.00
1½"x2" - CC-V840 Line No. 798-45 SC 2	CC-20213	9.61	18.00	2.51	18.00	16.71	27.00
1½"x2" - CC-V474 Line No. 827-20 SC 2	CC-20213	11.19	18.00	2.31	18.00	20.08	27.00
3"x4" - CC-V120 Line No. 827-27 SC 3	CC-20214	14.73	18.00	2.14	18.00	23.28	27.00
1½"x2½" - CC-V435 Line No. 712-4 SC 3	CC-20209	14.62	18.00	6.00	18.00	23.23	27.00
1½"x2" - CC-V363 Line No. 785-7 SC 3	CC-20209	4.04	18.00	1.79	18.00	6.30	27.00

PIPING SYSTEM	**SYSTEM DRAWING NO.**	**NORMAL & UPSET CONDITION**		**EMERGENCY CONDITION**		**FAULTED CONDITION**	
		ACTUAL (KSI)	**ACTUAL (KSI)**	**ACTUAL (KSI)**	**ACTUAL (KSI)**	**ACTUAL (KSI)**	**ACTUAL (KSI)**
1½x2" - CC-V235 **Line No. 709-7** **SC 2**	CC-20215	3.68	18.00	1.66	18.00	5.70	27.00
1½"x2" -2-CC-V408 **Line No. 790-9** **SC 3**	CC-20209	7.86	18.00	6.33	18.00	9.39	27.00
3/4"x1" - CC-V30 **Line No. 758-5** **SC 3**	CC-20205	8.45	18.00	0.35	18.00	16.55	27.00
2"x3" - CC-V343 **Line No. 816-5** **SC 3**	CC-20206	0.98	18.00	0.77	18.00	1.25	27.00
3/4"x1" - CC-V342 **Line No. 815-3** **SC 3**	CC-20206	2.68	18.00	1.63	18.00	3.73	27.00
1½"x2" - CC-V845 **Line No. 752-25** **SC 2**	CC-20207	10.91	18.00	2.31	18.00	19.51	27.00

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
1½"x2" - CC-V410 **Line No. 777-13** **SC 2**	CC-20207	8.57	18.00	2.31	18.00	14.84	27.00
1½"x2½" - CC-V407 **Line No. 782-6** **SC 3**	CC-20207	5.21	18.00	0.66	18.00	9.78	27.00
3/4"x1" - CC-V135 **Line No. 781-7** **SC 3**	CC-20207	15.59	18.00	5.11	18.00	26.08	27.00
3/4"x1" - CC-V141 **Line No. 783-4** **SC 3**	CC-20207	4.40	18.00	1.52	18.00	7.27	27.00
3/4"-1" - CC-V26 **Line No. 759-4** **SC 3**	CC-20207	1.95	18.00	0.38	18.00	3.5	27.00
3"x4" - CC-V143 **Line No. 784-5** **SC 3**	CC-20207	0.91	18.00	0.74	18.00	1.07	27.00

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
1½"x2½ - CC-V409 **Line No. 793-12** **SC 3**	CC-20205	3.72	18.00	2.07	18.00	5.37	27.00
1½"x2" - CC-V647 **Line No. 795-7** **SC 3**	CC-20205	16.45	18.00	3.69	18.00	19.14	27.00
1½"X2½" - CC-V264 **Line No. 833-3** **SC 3**	CC-20213	6.23	18.00	0.66	18.00	11.8	27.00
3/4"-1" - CC-V262 **Line No. 832-2** **SC 3**	CC-20213	4.04	18.00	1.85	18.00	6.23	27.00
3/4"-1" - CC-V269 **Line No. 834-4** **SC 3**	CC-20213	3.31	18.00	1.05	18.00	5.58	27.00
3/4"-1" - CC-V322 **Line No. 861-4** **SC 3**	CC-20213	1.93	18.00	0.38	18.00	3.49	27.00

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
3/4"-4" - CC-V271 **Line No. 835-5** **SC 3**	CC-20213	0.96	18.00	0.74	18.00	1.17	27.00
1½"x2½" - CC-V442 **Line No. 837-1** **SC 3**	CC-20211	5.27	18.00	1.94	18.00	8.61	27.00
6½"x2" - CC-V171 **Line No. 794-6** **SC 3**	CC-20212	14.42	18.00	3.69	18.00	22.22	27.00
SERVICE WATER SYSTEM							
2"x3" - SW-V32 **Line No. 1810-6** **SC 3**	SW-20795	9.14	18.00	2.05	18.00	16.30	27.00
3/4"x3/4" - SW-V514-A **Line No. 1834-5** **SC 3**	SW-20795	4.75	9.72	2.13	9.72	7.38	14.58

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
3/4"x3/4" - SW-V514-B	SW-20795	8.45	9.72	2.59	9.72	14.3	14.58
Line No. 1836-5							
SC 3							
2"x3" - SW-V73	SW-20795	6.28	18.00	1.42	18.00	11.14	27.00
Line No. 1812-6							
SC 3							
1½"x2" - SW-V214	SW-20794	2.28	18.00	0.78	18.00	3.79	27.00
Line No. 1846-7							
SC 3							
DEMINERALIZED WATER SYSTEM							
1½"x2" - DM-V18	DM-20352	4.74	19.78	1.15	19.78	16.45	29.67
Line No. 1608-6							
SC 2							

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
REACTOR COOLANT DRAIN TANK							
1½"x2" - WLD-V213	WLD-20218	12.26	19.68	3.17	19.68	21.35	29.52
Line No. 2102-12							
SC 2							
MAIN STEAM							
4000-41 (30")	MS-20580	19.88	21.00	10.91	21.00	25.36	31.50
4000-41 (30")	MS-20583	19.88	21.00	10.91	21.00	25.36	31.50
4001-41 (30")	MS-20581	19.88	21.00	10.91	21.00	25.36	31.50
4001-41 (30")	MS-20583	19.88	21.00	10.91	21.00	25.36	31.50
4002-37 (30")	MS-20581	19.88	21.00	10.91	21.00	25.36	31.50
4002-37 (30")	MS-20583	19.88	21.00	10.91	21.00	25.36	31.50
4003-37 (30")	MS-20580	19.88	21.00	10.91	21.00	25.36	31.50
4003-37 (30")	MS-20583	19.88	21.00	10.91	21.00	25.36	31.50
4000-11 (6")	MS-20580	11.52	18.00	7.72	18.00	20.54	27.00
4001-11 (6")	MS-20581	11.52	18.00	7.72	18.00	20.54	27.00
4002-09 (6")	MS-20581	11.03	18.00	7.65	18.00	17.59	27.00
4003-09 (6")	MS-20580	11.03	18.00	7.65	18.00	17.59	27.00

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
4000-15 (10")	MS-20580	1.33	18.00	1.33	18.00	1.33	27.00
4000-19	MS-20580	1.33	18.00	1.33	18.00	1.33	27.00
4000-23	MS-20580	1.33	18.00	1.33	18.00	1.33	27.00
4000-27	MS-20580	1.33	18.00	1.33	18.00	1.33	27.00
4000-30	MS-20580	1.33	18.00	1.33	18.00	1.33	27.00
4001-15	MS-20581	1.33	18.00	1.33	18.00	1.33	27.00
4001-19	MS-20581	1.33	18.00	1.33	18.00	1.33	27.00
4001-23	MS-20581	1.33	18.00	1.33	18.00	1.33	27.00
4001-27	MS-20581	1.33	18.00	1.33	18.00	1.33	27.00
4001-30	MS-20581	1.33	18.00	1.33	18.00	1.33	27.00
4002-13	MS-20581	1.33	18.00	1.33	18.00	1.33	27.00
4002-17	MS-20581	1.33	18.00	1.33	18.00	1.33	27.00
4002-21	MS-20581	1.33	18.00	1.33	18.00	1.33	27.00
4002-25	MS-20581	1.33	18.00	1.33	18.00	1.33	27.00
4002-28	MS-20581	1.33	18.00	1.33	18.00	1.33	27.00
4003-13	MS-20580	1.33	18.00	1.33	18.00	1.33	27.00
4003-17	MS-20580	1.33	18.00	1.33	18.00	1.33	27.00
4003-21	MS-20580	1.33	18.00	1.33	18.00	1.33	27.00

PIPING SYSTEM	SYSTEM DRAWING NO.	NORMAL & UPSET CONDITION		EMERGENCY CONDITION		FAULTED CONDITION	
		ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)	ACTUAL (KSI)
4003-25	MS-20580	1.33	18.00	1.33	18.00	1.33	27.00
4003-28	MS-20580	1.33	18.00	1.33	18.00	1.33	27.00

TABLE 3.9(B)-23 ASME CODE CLASS 1, 2 AND 3 PIPE SUPPORT LOAD COMBINATIONS

Normal & Upset	Faulted
D, T, OBE, TR,	D, T, SSE, TR, SAD (SSE), PAD,
SAD (OBE)	TAD, LOCA displacement

Definitions of Acronyms Used Above

Symbols for Stress Classification and Stress Limits are in accordance with ASME Section III. Other load symbols and definitions are specified below:

D - Deadweight, consists of the weight of the pipe and pipe supported elements such as valves and flanges, including weight of insulation and contained fluid.

T - Thermal loads due to:

a. Piping thermal expansion when subjected to maximum temperature difference between the fluid and the surrounding environment in the specified plant conditions, and

b. Anchor displacement due to thermal movements of piping anchors

TR - Thrust or transient due to safety valve discharge, valve trip or fluid flow

SAD - Seismic anchor displacement (OBE or SSE), affects piping supported from different structures of relative seismic motions

PAD - Anchor displacement due to pressure, e.g., containment building penetrations due to internal pressure during test or LOCA

TAD - Anchor displacement due to thermal growth of the structure e.g., radial and vertical growth of Containment Building during LOCA ± MSL-Dynamic System Load, Accident Load affecting piping as follows:

- Impact from missiles or pipe whip
- Jet impingement
- External pressure

OBE - Loads generated by the Operating Basis Earthquake (OBE), which is the earthquake that could reasonably be expected to affect the plant site during the operating life of the plant and which produces the vibratory ground motion for which those features of the nuclear plant necessary for continued operation without undue risk to the health and safety of the public have been designed to remain functional.

SSE - Loads generated by the Safe Shutdown Earthquake (SSE) which is the earthquake that produces the maximum vibratory ground motion for which certain structures, systems, and components important to safety and required for safe shutdown on the plant have been designed to remain functional.

TABLE 3.9(B)-26 ACTIVE PUMPS

Pump	Item No.	System	ASME Safety Class	Normal Mode	Post Loca Mode	Function
PCCW Pumps	CC-11A, B, C, D	CC	3	ON	ON	Primary Component Cooling Water
SW Pumps	SW-P-41A, B, C, D	SW	3	ON	ON	Ultimate Heat Sink (Ocean)
Cooling Tower Pumps	SW-P-110A, B	SW	3	OFF	ON	Ultimate Heat Sink (Tower)
EFW Pumps	FW-P-37A, B	FW	3	OFF	ON	Emergency Feed Water
Diesel Generator Fuel Oil Pumps*	DG-P-38A, B	DG	3	OFF	ON	Day Tank Fuel Supply
Containment Spray Pump	CBS-P-9A, B	CBS	2	OFF	ON	Containment Building Spray
Spent Fuel Pool Cooling Pumps*	SF-P-10A, B, C	SF	3	ON	ON	Core Residual Heat Removal

* Tested periodically by Station procedures, but not included in the IST program.

TABLE 3.9(B)-27 ACTIVE VALVE LIST BY SYSTEM

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
AS	V-175	Gate	3	12.00	Motor
AS	V-176	Gate	3	12.00	Motor
CC	MM-762	Relief [(1)]	3	10.00	Self
CC	MM-763	Relief [(1)]	3	10.00	Self
CC	V-1	Check	3	24.00	Delta P
CC	V-4	Check	3	24.00	Delta P
CC	V-32	Butt	3	10.00	Air
CC	V-57	Butt	2	12.00	Air
CC	V-121	Butt	2	12.00	Air
CC	V-122	Butt	2	12.00	Air
CC	V-137	Butt	3	14.00	Motor
CC	V-145	Butt	3	16.00	Motor
CC	V-168	Butt	2	12.00	Air
CC	V-175	Butt	2	12.00	Air
CC	V-176	Butt	2	12.00	Air
CC	V-256	Butt	2	12.00	Air
CC	V-257	Butt	2	12.00	Air
CC	V-266	Butt	3	14.00	Motor
CC	V-272	Butt	3	16.00	Motor
CC	V-295	Check	3	24.00	Delta P

[(1)] These devices are rupture discs and are classified as valves in the ASME Code. However, end loading calculations do not apply.

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
CC	V-298	Check	3	24.00	Delta P
CC	V-341	Butt	3	6.00	Air
CC	V-410	Relief	2	1.50	Self
CC	V-426	Butt	3	20.00	Air
CC	V-427	Butt	3	20.00	Air
CC	V-445	Butt	3	10.00	Air
CC	V-447	Butt	3	20.00	Air
CC	V-448	Butt	3	20.00	Air
CC	V-474	Relief	2	1.50	Self
CC	V-840	Relief	2	1.50	Self
CC	V-845	Relief	2	1.50	Self
CC	V-1298	Ball	3	1.00	Air
CC	V-1301	Ball	3	1.00	Air
CC	V-975	Ball	3	1.00	Air
CC	V-986	Ball	3	1.00	Air
CC	V-1105	Relief	2	0.75	Self
CC	V-1112	Relief	2	0.75	Self
CC	V-1277	Relief	3	0.75	Self
CC	V-1278	Relief	3	0.75	Self
CC	V-1279	Relief	3	0.75	Self
CC	V-1282	Check	3	2.00	Delta P
CC	V-1283	Check	3	2.00	Delta P
CC	TV-2171-1	Butt	3	24.00	Air
CC	TV-2171-2	Butt	3	24.00	Air
CC	TV-2271-1	Butt	3	24.00	Air
CC	TV-2271-2	Butt	3	24.00	Air

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
CO	V-142	Gate	3	24.00	Manual
CO	V-421	Check	3	2.00	Delta P
CO	V-422	Check	3	2.00	Delta P
CS	V-2	Check	1	2.00	Delta P
CS	V-4	Check	2	2.00	Delta P
CS	V-18	Check	1	2.00	Delta P
CS	V-20	Check	2	2.00	Delta P
CS	V-34	Check	1	2.00	Delta P
CS	V-36	Check	2	2.00	Delta P
CS	V-50	Check	1	2.00	Delta P
CS	V-52	Check	2	2.00	Delta P
CS	V-148	Relief	2	2.00	Self
CS	V-199	Check	2	2.00	Delta P
CS	V-211	Check	2	2.00	Delta P
CS	V-410	Gate	3	4.00	Manual
CS	V-416	Gate	3	4.00	Manual
CS	V-423	Weir	3	2.00	Manual
CS	V-424	Weir	3	2.00	Manual
CS	V-427	Check	1	2.00	Delta P
CS	V-430	Weir	3	2.00	Manual
CS	V-431	Weir	3	2.00	Manual
CS	V-442	Gate	2	4.00	Manual
CS	V-449	Check	3	2.00	Delta P
CS	V-453	Check	3	2.00	Delta P
CS	V-471	Check	1	2.00	Delta P
CS	V-472	Check	1	2.00	Delta P
CS	V-473	Check	1	2.00	Delta P
CS	V-474	Check	1	2.00	Delta P

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
CS	V-794	Relief	2	0.75	Self
CS	V-828 (2)	Gate	2	3.00	Manual
CS	V-829 (2)	Gate	2	3.00	Manual
DG	A FY-AS1 (3)	Three-way	3	0.375	Solen
DG	A FY-AS2 (3)	Three-way	3	0.375	Solen
DG	TCV7A-1 (3)	Three-way	3	6.00	Air
DG	TCV7A-2 (3)	Three-way	3	6.00	Air
DG	V-2A (3)	Check	3	6.00	Delta P
DG	V-5A (3)	Check	3	6.00	Delta P
DG	V-23A (3)	Check	3	5.00	Delta P
DG	V-24A (3)	Check	3	5.00	Delta P
DG	V-29A (3)	Three-way	3	5.00	Self
DG	V-62A	Relief	3	0.75	Self
DG	V-66A	Relief	3	0.75	Self
DG	V-69A	Check	3	0.75	Delta P
DG	V-70A	Check	3	0.75	Delta P
DG	V-84A (3)	Check	3	1.00	Delta P
DG	B-FY-AS1 (3)	Three-way	3	0.375	Solen
DG	B-FY-AS2 (3)	Three-way	3	0.375	Solen
DG	TCV7B-1 (3)	Three-way	3	6.00	Air
DG	TCV7B-2 (3)	Three-way	3	6.00	Air
DG	V-2B (3)	Check	3	6.00	Delta P
DG	V-5B (3)	Check	3	6.00	Delta P
DG	V-23B (3)	Check	3	5.00	Delta P
DG	V-24B (3)	Check	3	5.00	Delta P
DG	V-29B (3)	Three-way	3	5.00	Self

(2) This valve may require repositioning only during shutdown beyond hot standby. As such, it is not required to be included in the IST Program; it may be periodically tested by another station procedure.

(3) This is a skid-mounted component that is excluded from the IST Program and is periodically tested by another Station procedure.

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
DG	V-62B	Relief	3	0.75	Self
DG	V-66B	Relief	3	0.75	Self
DG	V-69B	Check	3	0.75	Delta P
DG	V-70B	Check	3	0.75	Delta P
DG	V-84B [(3)]	Check	3	1.00	Delta P
DG	V-115	Check	3	1.50	Delta P
DG	V-121	Check	3	1.50	Delta P
DM	V-18	Relief	2	1.50	Self
DM	V-611	Check	3	6.00	Delta P
DM	V-612	Check	3	6.00	Delta P
FW	V-30	Gate	2	18.00	Air
FW	V-39	Gate	2	18.00	Air
FW	V-48	Gate	2	18.00	Air
FW	V-57	Gate	2	18.00	Air
FW	V-64	Check	3	6.00	Delta P
FW	V-70	Check	3	6.00	Delta P
FW	V-76	Schk	2	4.00	Delta P
FW	V-82	Schk	2	4.00	Delta P
FW	V-88	Schk	2	4.00	Delta P
FW	V-94	Schk	2	4.00	Delta P
FW	V-154	Globe	2	4.00	Manual
FW	V-216	Schk	3	6.00	Delta P
FW	V-330	Check	2	18.00	Delta P
FW	V-331	Check	2	18.00	Delta P
FW	V-332	Check	2	18.00	Delta P
FW	V-333	Check	2	18.00	Delta P

(3) This is a skid-mounted component that is excluded from the IST Program and is periodically tested by another Station procedure.

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
FW	FV-4214-A	Globe	3	4.00	Motor
FW	FV-4214-B	Globe	3	4.00	Motor
FW	FV-4224-A	Globe	3	4.00	Motor
FW	FV-4224-B	Globe	3	4.00	Motor
FW	FV-4234-A	Globe	3	4.00	Motor
FW	FV-4234-B	Globe	3	4.00	Motor
FW	FV-4244-A	Globe	3	4.00	Motor
FW	FV-4244-B	Globe	3	4.00	Motor
FW	V-357	Check	3	6.00	Delta P
FW	V-346	Globe	3	4.00	Motor
FW	V-347	Globe	3	4.00	Motor
FW	V-349	Check	3	4.00	Delta P
FW	V-350	Check	3	3.00	Delta P
FW	V-353	Check	3	3.00	Delta P
FW	V-351	Check	3	1.00	Delta P
IA	V-531	Check	2	2.00	Delta P
MS	V-5	Gate	2	6.00	Manual
MS	V-6	Relief	2	6.00	Self
MS	V-7	Relief	2	6.00	Self
MS	V-8	Relief	2	6.00	Self
MS	V-9	Relief	2	6.00	Self
MS	V-10	Relief	2	6.00	Self
MS	V-21	Gate	2	6.00	Manual
MS	V-22	Relief	2	6.00	Self
MS	V-23	Relief	2	6.00	Self
MS	V-24	Relief	2	6.00	Self
MS	V-25	Relief	2	6.00	Self

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
MS	V-26	Relief	2	6.00	Self
MS	V-35	Gate	2	6.00	Manual
MS	V-36	Relief	2	6.00	Self
MS	V-37	Relief	2	6.00	Self
MS	V-38	Relief	2	6.00	Self
MS	V-39	Relief	2	6.00	Self
MS	V-40	Relief	2	6.00	Self
MS	V-49	Gate	2	6.00	Manual
MS	V-50	Relief	2	6.00	Self
MS	V-51	Relief	2	6.00	Self
MS	V-52	Relief	2	6.00	Self
MS	V-53	Relief	2	6.00	Self
MS	V-54	Relief	2	6.00	Self
MS	V-86	Gate	2	30.00	Phyd
MS	V-88	Gate	2	30.00	Phyd
MS	V-90	Gate	2	30.00	Phyd
MS	V-92	Gate	2	30.00	Phyd
MS	V-94	Check	3	4.00	Delta P
MS	V-96	Check	3	4.00	Delta P
MS	V-204	Globe	2	4.00	Motor
MS	V-205	Globe	2	4.00	Motor
MS	V-206	Globe	2	4.00	Motor
MS	V-207	Globe	2	4.00	Motor
MS	PV-3001	Globe	2	10.00	Air
MS	PV-3002	Globe	2	10.00	Air
MS	PV-3003	Globe	2	10.00	Air
MS	PV-3004	Globe	2	10.00	Air

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
MS	V-400	Check	3	0.75	Delta P
MS	V-401	Check	3	0.75	Delta P
MS	V-404	Check	3	0.75	Delta P
MS	V-405	Check	3	0.75	Delta P
MS	V-417	Check	3	0.75	Delta P
MS	V-418	Check	3	0.75	Delta P
MS	V-393	Globe	2	3.00	Air
MS	V-394	Globe	2	3.00	Air
MS	V-395	Globe	3	4.00	Air
NG	V-14	Globe	2	1.00	Air
NG	FV-4609	Globe	2	1.00	Solen
NG	FV-4610	Globe	2	1.00	Solen
RC	V-312	Relief	2	0.75	Self
RC	V-314	Relief	2	0.75	Self
RC	V-323	Globe	2	0.75	Motor
RC	V-337	Relief	2	0.75	Self
RC	V-360	Relief	2	0.75	Self
RC	V-361	Relief	2	0.75	Self
RC	V-475	Check	2	0.50	Delta P
RC	V-479	Check	2	0.50	Delta P
RC	FV-2830	Globe	2	0.50	Solen
RC	FV-2831	Globe	2	0.50	Solen
RC	FV-2832	Globe	2	0.50	Solen
RC	FV-2833	Globe	2	0.50	Solen
RC	FV-2836	Globe	2	0.50	Solen
RC	FV-2837	Globe	2	0.50	Solen
RC	FV-2840	Globe	2	0.50	Solen

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
RC	FV-2874	Globe	2	0.50	Solen
RC	FV-2876	Globe	2	0.50	Solen
RC	FV-2881	Globe	2	0.75	Solen
RC	FV-2894	Globe	2	0.50	Solen
RC	FV-2896	Globe	2	0.50	Solen
RH	V-16	Globe	2	0.75	Air
RH	V-17	Globe	2	0.75	Air
RH	V-18 (2)	Globe	2	2.00	Manual
RH	V-19 (2)	Globe	2	2.00	Manual
SB	V-1	Gate	2	3.00	Air
SB	V-3	Gate	2	3.00	Air
SB	V-5	Gate	2	3.00	Air
SB	V-7	Gate	2	3.00	Air
SB	V-9	Gate	2	3.00	Air
SB	V-10	Gate	2	3.00	Air
SB	V-11	Gate	2	3.00	Air
SB	V-12	Gate	2	3.00	Air
SF	V-3 (4)	Check	3	6.00	Delta P
SF	V-7 (4)	Check	3	6.00	Delta P
SF	V-101	Relief	2	0.75	Self
SI	V-81	Check	1	2.00	Delta P
SI	V-86	Check	1	2.00	Delta P
SI	V-88	Check	2	1.50	Delta P
SI	V-91	Check	2	1.50	Delta P
SI	V-106	Check	1	2.00	Delta P
SI	V-110	Check	1	2.00	Delta P

(2) This valve may require repositioning only during shutdown beyond hot standby. As such, it is not required to be included in the IST Program; it may be periodically tested by another station procedure.

(4) Tested periodically by Station procedures, but not included in the IST Program because spent fuel pool cooling function does not meet ASME OM Code criteria.

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
SI	V-118	Check	1	2.00	Delta P
SI	V-122	Check	1	2.00	Delta P
SI	V-126	Check	1	2.00	Delta P
SI	V-130	Check	1	2.00	Delta P
SI	V-144	Check	1	1.50	Delta P
SI	V-148	Check	1	1.50	Delta P
SI	V-152	Check	1	1.50	Delta P
SI	V-156	Check	1	1.50	Delta P
SI	V-247	Relief	2	0.75	Self
SI	FV-2475 [5]	Globe	2	1.00	Solen
SI	FV-2476 [5]	Globe	2	1.00	Solen
SI	FV-2477 [5]	Globe	2	1.00	Solen
SI	FV-2482 [5]	Globe	2	1.00	Solen
SI	FV-2483 [5]	Globe	2	1.00	Solen
SI	FV-2486 [5]	Globe	2	1.00	Solen
SI	FV-2495 [5]	Globe	2	1.00	Solen
SI	FV-2496 [5]	Globe	2	1.00	Solen
RH	FCV-610	Globe	2	3.00	Motor
RH	FCV-611	Globe	2	3.00	Motor
SS	V-273	Check	2	0.50	Delta P
SS	FV-2857	Globe	2	0.50	Solen
SW	V-1	Check	3	24.00	Delta P
SW	V-2	Butt	3	24.00	Motor
SW	V-3	Check	3	24.00	Delta P

[5] This valve may be required to be repositioned during a plant cooldown beyond hot standby and is required to remain in its normal position during Modes 1, 2, and 3. The normal passive function must be included in the IST Program. Its active function may be periodically tested by another Station procedure.

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
SW	V-4	Butt	3	12.00	Motor
SW	V-5	Butt	3	12.00	Motor
SW	V-16	Butt	3	16.00	Air
SW	V-18	Butt	3	16.00	Air
SW	V-19	Butt	3	24.00	Motor
SW	V-20	Butt	3	24.00	Motor
SW	V-22	Butt	3	24.00	Motor
SW	V-23	Butt	3	24.00	Motor
SW	V-24	Check	3	24.00	Delta P
SW	V-25	Butt	3	24.00	Motor
SW	V-27	Butt	3	24.00	Motor
SW	V-28	Check	3	24.00	Delta P
SW	V-29	Butt	3	24.00	Motor
SW	V-30	Check	3	24.00	Delta P
SW	V-31	Butt	3	24.00	Motor
SW	V-34	Butt	3	24.00	Motor
SW	V-53	Check	3	24.00	Delta P
SW	V-54	Butt	3	24.00	Motor
SW	V-56	Butt	3	24.00	Motor
SW	V-74	Butt	3	24.00	Motor
SW	V-76	Butt	3	24.00	Motor
SW	V-139	Butt	3	24.00	Motor
SW	V-140	Butt	3	24.00	Motor

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
SW	V-174	Check	3	1.00	Delta P
SW	V-175	Check	3	1.00	Delta P
SW	V-176	Check	3	1.00	Delta P
SW	V-177	Check	3	1.00	Delta P
VG	FV-1661	Globe	2	2.00	Solen
VG	FV-1712	Globe	2	2.00	Solen
CAH	FV-6572	Gate	2	0.50	Solen
CAH	FV-6573	Gate	2	0.50	Solen
CAH	FV-6574	Gate	2	0.50	Solen
CAH	V-12	Check	2	0.50	Delta P
CAP	V-1	Butt	2	36.00	Air
CAP	V-2	Butt	2	36.00	Air
CAP	V-3	Butt	2	36.00	Air
CAP	V-4	Butt	2	36.00	Air
CBS	V-3	Check	2	12.00	Delta P
CBS	V-7	Check	2	12.00	Delta P
CBS	V-8	Gate	2	16.00	Motor
CBS	V-9	Check	2	12.00	Delta P
CBS	V-11	Gate	2	8.00	Motor
CBS	V-12	Check	2	8.00	Delta P
CBS	V-14	Gate	2	16.00	Motor
CBS	V-15	Check	2	12.00	Delta P
CBS	V-17	Gate	2	8.00	Motor
CBS	V-18	Check	2	8.00	Delta P
CBS	V-25	Check	2	16.00	Delta P

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
CBS	V-26	Check	2	16.00	Delta P
CBS	V-31	Butt	2	4.00	Air
CBS	V-32	Butt	2	4.00	Air
CBS	V-33	Butt	2	4.00	Air
CBS	V-38	Gate	2	6.00	Motor
CBS	V-43	Gate	2	6.00	Motor
CBS	V-94	Relief	2	0.75	Self
CBS	V-96	Relief	2	0.75	Self
CBS	V-147	Check	2	16.00	Delta P
CBS	V-148	Check	2	16.00	Delta P
CBS	V-149 (6)	Relief	2	0.75	Self
CBS	V-150 (6)	Relief	2	0.75	Self
CBS	V-151 (6)	Relief	2	0.75	Self
CBS	V-152 (6)	Relief	2	0.75	Self
CGC	V-3	Globe	2	1.00	Manual
CGC	V-4	Check	2	1.00	Delta P
CGC	V-10	Globe	2	1.00	Manual
CGC	V-12	Globe	2	1.00	Manual
CGC	V-13	Globe	2	1.00	Manual
CGC	V-14	Globe	2	2.00	Motor
CGC	V-24	Globe	2	1.00	Manual
CGC	V-25	Check	2	1.00	Delta P
CGC	V-28	Globe	2	2.00	Motor
CGC	V-32	Globe	2	1.00	Manual
CGC	V-34	Globe	2	1.00	Manual
CGC	V-35	Globe	2	1.00	Manual

(6) This valve is required to perform its active safety function only during shutdown modes beyond hot standby. It is still included in the IST Program per the requirements of the ASME OM Code.

SYS	TAG NO	TYPE	SAFETY CLASS	SIZE	ACTUATED BY
COP	V-1	Butt	2	8.00	Air
COP	V-2	Butt	2	8.00	Air
COP	V-3	Butt	2	8.00	Air
COP	V-4	Butt	2	8.00	Air
MSD	V-44	Globe	2	1.00	Motor
MSD	V-45	Globe	2	1.00	Motor
MSD	V-46	Globe	2	1.00	Motor
MSD	V-47	Globe	2	1.00	Motor
RMW	V-30	Globe	2	3.00	Air
RMW	V-31	Weir	3	2.00	Manual
RMW	V-34	Weir	3	2.00	Manual
RMW	V-119	Check	2	2.00	Delta P
WLD	V-81	Globe	2	3.00	Air
WLD	V-82	Globe	2	3.00	Air
WLD	V-209	Relief	2	0.75	Self
WLD	V-213	Relief	2	1.50	Self
WLD	FV-8330	Globe	2	2.00	Solen
WLD	FV-8331	Globe	2	2.00	Solen

LEGEND

VALVE-TYPE

BUTT - Butterfly

SCHK - Stop-Check

ACTUATED-BY

SOLEN - Solenoid

PHYD - Pneumatic-Hydraulic

DELTA P - Delta Pressure

TABLE 3.9(B)-28 VALVES THAT PERFORM A MECHANICAL MOTION TO ACCOMPLISH OR SUPPORT A SAFETY FUNCTION (NON-ASME III OR NON-1E)

Tag No.	Type	ANS Safety Class	Actuated By	Normal Position
CBA-TCV-21200A	Three-way	3	Motor	Open
CBA-TCV-21200B	Three-way	3	Motor	Open
CC-V-120	Relief	NNS	Self	Closed
CC-V-486	Relief	NNS	Self	Closed
CS-V-834	Relief	NNS	Self	Closed
DG-V-18A	Check	3	Delta P	Closed
DG-V-31A	Check	3	Delta P	Closed
DG-V-41A	Gate	3	Self	Open
DG-V-52A	Regulator	3	Self	Open
DG-V-54A	Relief	3	Self	Closed
DG-V-59A	Shuttle	3	Self	Closed
DG-V-60A	Control	3	Self	Closed
DG-V-72A	Relief	3	Self	Closed
DG-V-82A	Check	3	Delta P	Closed
DG-V-83A	Check	3	Delta P	Closed
DG-V-85A	Check	3	Delta P	Closed
DG-V-87A	Check	3	Delta P	Open
DG-V-94A	Check	3	Delta P	Open
DG-V-100A	Relief	3	Self	Closed
DG-V-196A	Relief	3	Self	Closed
DG-V-220A	Shuttle	3	Self	Closed
DG-V-221A	Shuttle	3	Self	Closed
DG-V-224A	Control	3	Self	Closed
DG-V-225A	Gate	3	Manual	Open
DG-V-253A	Three-way	3	Solenoid	Closed
DG-V-257A	Regulator	3	Self	Closed

Tag No.	Type	ANS Safety Class	Actuated By	Normal Position
DG-V-258A	Relief	3	Self	Closed
DG-V-259A	Relief	3	Self	Closed
DG-V-260A	Check	3	Delta P	Closed
DG-V-261A	Check	3	Delta P	Closed
DG-V-269A	Shuttle	3	Self	Closed
DG-V-279A	Globe	3	Solenoid	Open
DG-V-280A	Globe	3	Solenoid	Closed
DG-V-281A	Check	3	Delta P	Closed
DG-V-282A	Check	3	Delta P	Closed
DG-V-285A	Globe	3	Solenoid	Closed
DG-V-286A	Check	3	Delta P	Closed
DG-V-287A	Check	3	Delta P	Closed
DG-V-288A	Globe	3	Solenoid	Closed
DG-V-289A	Globe	3	Solenoid	Closed
DG-V-325A	Gate	3	Solenoid	Open
DG-V-331A	Relief	3	Self	Closed
DG-V-332A	Ball	3	Manual	Closed
DG-V-333A	Ball	3	Manual	Closed
DG-V-334A	Check	3	Delta P	Closed
DG-V-335A	Relief	3	Self	Closed
DG-V-18B	Check	3	Delta P	Closed
DG-V-31B	Check	3	Delta P	Closed
DG-V-41B	Gate	3	Self	Open
DG-V-52B	Regulator	3	Self	Open
DG-V-54B	Relief	3	Self	Closed
DG-V-59B	Shuttle	3	Self	Closed
DG-V-60B	Control	3	Self	Closed
DG-V-72B	Relief	3	Self	Closed
DG-V-82B	Check	3	Delta P	Closed

Tag No.	Type	ANS Safety Class	Actuated By	Normal Position
DG-V-83B	Check	3	Delta P	Closed
DG-V-85B	Check	3	Delta P	Closed
DG-V-87B	Check	3	Delta P	Open
DG-V-94B	Check	3	Delta P	Open
DG-V-100B	Relief	3	Self	Closed
DG-V-196B	Relief	3	Self	Closed
DG-V-220B	Shuttle	3	Self	Closed
DG-V-221B	Shuttle	3	Self	Closed
DG-V-224B	Control	3	Self	Closed
DG-V-225B	Gate	3	Manual	Open
DG-V-253B	Three-way	3	Solenoid	Closed
DG-V-257B	Regulator	3	Self	Closed
DG-V-258B	Relief	3	Self	Closed
DG-V-259B	Relief	3	Self	Closed
DG-V-260B	Check	3	Delta P	Closed
DG-V-261B	Check	3	Delta P	Closed
DG-V-269B	Shuttle	3	Self	Closed
DG-V-279B	Globe	3	Solenoid	Open
DG-V-280B	Globe	3	Solenoid	Closed
DG-V-281B	Check	3	Delta P	Closed
DG-V-282B	Check	3	Delta P	Closed
DG-V-285B	Globe	3	Solenoid	Closed
DG-V-286B	Check	3	Delta P	Closed
DG-V-287B	Check	3	Delta P	Closed
DG-V-288B	Globe	3	Solenoid	Closed
DG-V-289B	Globe	3	Solenoid	Closed
DG-V-325B	Gate	3	Solenoid	Open
DG-V-331B	Relief	3	Self	Closed
DG-V-332B	Ball	3	Manual	Closed
DG-V-333B	Ball	3	Manual	Closed
DG-V-334B	Check	3	Delta P	Closed
DG-V-335B	Relief	3	Self	Closed

Tag No.	Type	ANS Safety Class	Actuated By	Normal Position
DGA-PCV-9602-1	Regulator	3	Self	Open
DGA-PCV-9612-1	Regulator	3	Self	Open
DGB-PCV-9602-1	Regulator	3	Self	Open
DGB-PCV-9612-1	Regulator	3	Self	Open
DM-V-274	Relief	NNS	Self	Closed
FW-FCV-510	Globe	NNS	Air	Open
FW-FCV-520	Globe	NNS	Air	Open
FW-FCV-530	Globe	NNS	Air	Open
FW-FCV-540	Globe	NNS	Air	Open
FW-LV-4210	Globe	NNS	Air	Open
FW-LV-4220	Globe	NNS	Air	Open
FW-LV-4230	Globe	NNS	Air	Open
FW-LV-4240	Globe	NNS	Air	Open
FW-V-99	Check	NNS	Delta P	Closed
FW-V-163	Gate	NNS	Motor	Closed
IA-V-545	Check	3	Delta P	Open
IA-V-546	Check	3	Delta P	Open
IA-V-547	Check	3	Delta P	Open
IA-V-548	Check	3	Delta P	Open
IA-V-549	Check	3	Delta P	Open
IA-V-550	Check	3	Delta P	Open
IA-V-8030	Check	3	Delta P	Open
IA-V-8031	Check	3	Delta P	Open
IA-V-8032	Check	3	Delta P	Open
IA-V-8033	Check	3	Delta P	Open
RC-LCV-459	Globe	1	Air	Open
RC-LCV-460	Globe	1	Air	Open
RMW-V-107	Relief	NNS	Self	Closed
SF-V-214	Check	NNS	Delta P	Closed
SF-V-215	Check	NNS	Delta P	Closed
SI-V-314	Relief	NNS	Self	Closed
SW-V-179	Check	3	Delta P	Closed

Tag No.	Type	ANS Safety Class	Actuated By	Normal Position
SW-V-180	Check	3	Delta P	Closed
SW-V-224	Check	3	Delta P	Closed
SW-V-225	Check	3	Delta P	Closed
WLD-V-211	Relief	NNS	Self	Closed
WLD-V-277	Relief	NNS	Self	Closed

TABLE 3.9(B)-29 STRESS AND DEFLECTION ANALYSIS FOR THE SPENT FUEL POOL PUMPS (BINGHAM-WILLAMETTE CO. 6X8X12 CF)

Items – SSE Analysis	Actual	Allowable
Natural Frequency (hz)		
Pump	124	-
Motor	190	-
Pump Casing (psi)	2123	16500
Shaft (psi)	1317	28000
Pedestal Weld (psi)		
Pump	3593	32400
Motor	93	32400
Hold Down Bolts (psi)		
Pump	32035	70000
Motor	1179	70000
Base Hold Down Bolts (psi)		
Tension	5160	19100
Shear	1930	9900
Clearances (in.)		
Deflection @ coupling	0.0071	0.035
Deflection @ rotor	0.000662	0.0196*

* Air gap between rotor/stator

Table 3.9(N)-1 SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

Normal Conditions		Occurrences
1.	Heatup and cooldown at 100°F/hr (pressurizer cooldown 200°F/hr)	200 (each)
2.	Unit loading and unloading at 5% of full power/min	13,200 (each)
3.	Step load increase and decrease of 10% of full power	2,000 (each)
4.	Large step load decrease with steam dump	200
5.	Steady-state fluctuations : a. Initial fluctuations b. Random fluctuations	 $1.5x10^5$ $3.0x10^6$
6.	Feedwater cycling at hot shutdown	2000
7.	Loop out of service: a. Normal loop shutdown b. Normal loop startup	 80 70
8.	Feedwater Heaters Out of Service a. One Heater Out of Service b. One Bank of Heaters Out of Service	 120 120
9.	Unit loading and unloading between 0 to 15% of full power	500 (each)
10.	Boron concentration equalization	26,400
11.	Refueling	80
12.	Reduced temperature return to power	2000
13.	Reactor coolant pumps startup/shutdown	3800
14.	Turbine roll test	20
15.	Primary side leak test	200
16.	Secondary side leak test	80
17.	Tube leakage test	800
Upset Conditions		
1.	Loss of load, without immediate reactor trip	80
2.	Loss of power (blackout with natural circulation in the Reactor Coolant System)	40

3.	Partial loss of flow (loss of one pump)	80
4.	Reactor trip from full power	
	a. Without cooldown	230
	b. With cooldown, without safety injection	160
	c. With cooldown and safety injection	10
5.	Inadvertent reactor coolant depressurization	20
6.	Inadvertent startup of an inactive loop	10
7.	Control rod drop	80
8.	Inadvertent emergency core cooling system actuation	60
9.	Operating basic earthquake (5 earthquakes of 10 cycles each)	50
10.	Excessive feedwater flow	30
11.	RCS Cold Overpressurization	10
Emergency Condition*		
1.	Small loss-of-coolant accident	5
2.	Small steam line break	5
3.	Complete loss of flow	5
Faulted Conditions*		
1.	Main reactor coolant pipe break (large loss-of-coolant accident)	1
2.	Large steam line break	1
3.	Feedwater line break	1
4.	Reactor coolant pump locked rotor	1
5.	Control rod ejection	1
6.	Steam generator tube rupture	(included under upset conditions, reactor trip from full power with safety injection)
7.	Safe shutdown earthquake	1
Test Conditions		
1.	Primary side hydrostatic test	10
2.	Secondary side hydrostatic test	10
	*In accordance with the ASME Nuclear Power Plant Components Code, emergency and faulted conditions are not included in fatigue evaluation.	

TABLE 3.9(N)-2 LOADING COMBINATIONS FOR ASME CLASS 1 COMPONENTS AND SUPPORTS

Condition Classification	Loading Combination
Design	Design Pressure, Design Temperature, Deadweight, Operating Basis Earthquake[1]
Normal	Normal Condition Transients, Deadweight
Upset	Upset Condition Transients, Deadweight, Operating Basis Earthquake
Emergency	Emergency Condition Transients, Deadweight
Faulted	Faulted Condition transients, Deadweight, Safe Shutdown Earthquake or (Safe Shutdown earthquake and Pipe Rupture Loads)
The operating basis earthquake is evaluated to ASME Level B stress limits for Class 1 piping components.	

TABLE 3.9(N)-3 ALLOWABLE STRESSES FOR ASME SECTION III CLASS 1 COMPONENTS(1)

Operating Condition Classification	Vessels/Tanks	Piping	Pumps	Valves	Component Supports
Normal	ASME Section III	ASME Section III	ASME Section III	ASME Section III	ASME Section III Subsection NF
Upset	ASME Section III	ASME Section III	ASME Section III	ASME Section III	ASME Section III Subsection NF
Emergency	ASME Section III	ASME Section III	ASME Section III	ASME Section III	ASME Section III Subsection NF
Faulted	ASME Section III See Subsection 3.9(N).1.4	ASME Section III See Subsection 3.9(N).1.4	ASME Section III See Subsection 3.9(N).1.4 (No active Class 1 pump used)	Note (2).	ASME Section III Subsection NF See Subsection 3.9(N).1.4

P_e, P_m, P_b, Q_t, C_p, S_n & S_m as defined by Section III, ASME Code

Notes: (1) A test of the components may be performed in lieu of analysis.

(2) CLASS 1 VALVE FAULTED CONDITION CRITERIA

ACTIVE	INACTIVE
a) Calculate Pm from para. NB3545.1 with Internal Pressure Ps = 1.25P Pm ≤ 1.5Sm	a) Calculate Pm from para. NB3545.1 with Internal Pressure Ps = 1.50 Ps Pm ≤ 2.4Sm or 0.7 Su
b) Calculate Sn from para. NB3545.2 with	b) Calculate Sn from para. NB3545.2 with
Cp = 1.5 Ps = 1.25Ps	Cp = 1.5 Ps = 1.50Ps
Qt2 = 0 Ped = 1.3X value of Ped from equations of 3545.2(b)(1) Sn ≤ 3Sm	Qt2 = 0 Ped = 1.3X value of Ped from equations of 3545.2(b)(1) Sn ≤ 3Sm

TABLE 3.9(N)-4 DESIGN LOADING COMBINATIONS FOR ASME CODE CLASS 2 AND 3 COMPONENTS AND SUPPORTS[(1)]

Loading Combination[(2,3)]	Design/Service Level Requirements
1. Design pressureDesign temperature, Dead weight	Design
2. Normal condition pressure, normal condition metal temperature, deadweight, nozzle loads	Service Level A
3. Upset condition pressure, Upset condition metal temperature, deadweight, nozzle loads, Operating Basis Earthquake	Service Level B
4. Emergency condition pressure,emergency condition metal temperature, deadweight, nozzle loads	Service Level C
5. Faulted condition pressure, faulted condition metal temperature, deadweight, nozzle loads, Safe Shutdown Earthquake	Service Level D

(1) Excludes active pumps. Refer to Table 3.9(N)-8 for loading combinations and corresponding criteria applicable for active pumps.

(2) Temperature is used to determine allowable stress only.

(3) Nozzle loads, pressures, and temperatures are these associated with the respective plant operating conditions (i.e., normal, upset, emergency, and faulted) as noted, for the component under consideration.

TABLE 3.9(N)-5 STRESS CRITERIA FOR SAFETY-RELATED ASME CLASS 2 [(1)] AND CLASS 3 TANKS

Design/Service Level	Stress Limits
Design and Service Level A	$\sigma_m \leq 1.0$ S
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.5$ S
Service Level B	$\sigma_m \leq 1.1$ S
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65$ S
Service Level C	$\sigma_m \leq 1.5$ S
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80$ S
Service Level D	$\sigma_m \leq 2.0$ S
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4$ S

(1) applies for tanks designed in accordance with ASME III, NC-3300.

TABLE 3.9(N)-6 STRESS CRITERIA FOR SAFETY-RELATED CLASS 2 TANKS[1]

Design/Service Level	Stress Limits
Design and Service Level A	$P_m \leq 1.0\ S_m$
	$P_L \leq 1.5\ S_m$
	$(P_m \text{ or } P_L) + P_b \leq 1.5\ S_m$
Service Level B	$P_m \leq 1.1\ S_m$
	$P_L \leq 1.65\ S_m$
	$(P_m \text{ or } P_L) + P_b \leq 1.65\ S_m$
Service Level C	$P_m \leq 1.2\ S_m$
	$P_L \leq 1.8\ S_m$
	$(P_m \text{ or } P_L) + P_b \leq 1.8\ S_m$
Service Level D	$P_m \leq 2.0\ S_m$
	$P_L \leq 3.0\ S_m$
	$(P_m \text{ or } P_L) + P_b \leq 3.0\ S_m$

(1) Applies for tanks designed in accordance with ASME III, NC-3200.

TABLE 3.9(N)-7 STRESS CRITERIA FOR ASME CODE CLASS 2 AND 3 INACTIVE PUMPS

Design/Service Level	Stress Limits	P_{max}*
Design and Service Level A	$\sigma_m \leq 1.0$ S	
	$(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.5$ S	
Service Level B	$\sigma_m \leq 1.1$ S	1.1
	$(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S	
Service Level C	$\sigma_m \leq 1.5$ S	1.2
	$(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.80$ S	
Service Level D	$\sigma_m \leq 2.0$ S	1.5
	$(\sigma_m$ or $\sigma_L) + \sigma_b \leq 2.4$ S	

* The maximum pressure shall not exceed the tabulated factors listed under P_{max} times the design pressure.

TABLE 3.9(N)-8 DESIGN CRITERIA FOR ACTIVE PUMPS

Design/Service Level	Stress Limits
Design, Service Level A and Service Level B	$\sigma_m \leq 1.0$ S
	$\sigma_m + \sigma_b \leq 1.5$ S
Service Level C	$\sigma_m \leq 1.2$ S
	$\sigma_m + \sigma_b \leq 1.65$ S
Service Level D	$\sigma_m \leq 1.2$ S
	$\sigma_m + \sigma_b \leq 1.8$ S

TABLE 3.9(N)-9 STRESS CRITERIA FOR SAFETY-RELATED ASME CODE CLASS 2 AND CLASS 3 VALVES

Design/Service Level	Stress Limits (Notes 1-4)	P_{max} (Note 5)
Design & Service Level A	Valve bodies shall conform to ASME Section III	1.0
Service Level B	$\sigma_m \leq 1.1$ S	1.1
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65$ S	
Service Level C	$\sigma_m \leq 1.5$ S	1.2
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80$ S	
Service Level D	$\sigma_m \leq 2.0$ S	1.5
	$(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4$ S	

Notes:

1. Valve nozzle (piping load) stress analysis is not required when both the following conditions are satisfied: (1) the section modulus and area of energy plane, normal to the flow, through the region defined as the valve body crotch are at least 110% of those for the piping connected (or join-ed) to the valve body inlet and outlet nozzles; and, (2) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, for connected piping material. If the valve body material allowable stress is less than that of the connected piping, the required acceptance criteria ratio shall be 110% multiplied by the ratio of the piping allowable stress to the valve allowable stress. If unable to comply with this requirement, an analysis in accordance with the design procedure for Class 1 valves is an acceptable alternate method.
2. Casting quality factor of 1.0 shall be used.
3. These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.
4. Design requirements listed in this table are not applicable to valve discs, stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet, or otherwise not part of the pressure boundary.
5. The maximum pressure resulting from upset, emergency or faulted conditions shall not exceed the tabulated factors listed under P_{max} times the design pressure. If these pressure limits are met the stress limits in Table 3.9(N)-9 are considered to be satisfied.

TABLE 3.9(N)-10 **ACTIVE PUMPS**

Pump	Item No.	System	ASME Safety Class	Normal Mode	Post LOCA Mode	Function
Centrifugal charging pump No. 1	CS-P-2A	CVCS	2	ON/OFF	ON	High head safety injection
Centrifugal Charging pump No. 2	CS-P-2B	CVCS	2	ON/OFF	ON	High head safety injection
Boric acid transfer pumps Nos. 1 and 2	CS-P-3A CS-P-3B	BRS	2	ON/OFF	OFF	Boration and safe shutdown
Residual heat removal pumps Nos. 1 and 2	RH-P-8A RH-P-8B	RHRS	2	OFF	ON	Low head safety injection and normal cooldown
Safety injec-tion pumps Nos. 1 and 2	SI-P-6A SI-P-6B	SIS	2	OFF	ON	Safety injection

TABLE 3.9(N)-11 ACTIVE VALVES NSSS SUPPLIED

Valve Number	System	Actuated by	Size(in)	Type/ANS Safety Class
RC-V-24 [(1)]	RCS	Self	3	Relief/2
RC-V-89 [(1)]	RCS	Self	3	Relief/2
RC-V-115	RCS	Self Actuated	6	Safety/1
RC-V-116	RCS	Self Actuated	6	Safety/1
RC-V-117	RCS	Self Actuated	6	Safety/1
RC-V-122	RCS	Motor	4	Gate/1
RC-V-124	RCS	Motor	4	Gate/1
RC-PCV-456A	RCS	Solenoid	3x6	Relief/1
RC-PCV-456B	RCS	Solenoid	3x6	Relief/1
RMW-V-29	RCS	ΔP	3	Check/2
SF-V-197 [(3)]	SF	ΔP	8	Check/3
CS-V-142	CVCS	Motor	3	Gate/2
CS-V-143	CVCS	Motor	3	Gate/2
CS-V-144	CVCS	ΔP	3	Check/2
CS-V-149	CVCS	Motor	3	Gate/2
CS-V-150	CVCS	Air	3	Globe/2
CS-V-167	CVCS	Motor	2	Globe/2
CS-V-168	CVCS	Motor	2	Globe/2
CS-V-173	CVCS	Self	2	Relief/2
CS-V-178	CVCS	ΔP	3	Check/1
CS-V-179	CVCS	ΔP	3	Check/1

(1) This valve is required to perform its active safety function only during shutdown modes beyond hot standby. It is still included in the IST Program per the requirements of the ASME OM Code.

(3) Tested periodically by Station procedures, but not included in the IST program because spent fuel pool cooling function does not meet ASME OM Code criteria.

Valve Number	System	Actuated by	Size(in)	Type/ANS Safety Class
CS-V-181	CVCS	ΔP	3	Check/1
CS-V-182	CVCS	ΔP	3	Check/1
CS-V-192	CVCS	ΔP	4	Check/2
CS-V-196	CVCS	Motor	2	Globe/2
CS-V-197	CVCS	Motor	2	Globe/2
CS-V-200	CVCS	ΔP	4	Check/2
CS-V-209	CVCS	ΔP	4	Check/2
CS-V-210	CVCS	Manual	4	Gate/2
CS-V-219	CVCS	Manual	3	Globe/2
CS-V-220	CVCS	Manual	4	Gate/2
CS-V-221	CVCS	Manual	3	Globe/2
CS-V-213	CVCS	ΔP	3	Check/2
CS-V-227	CVCS	Self	3/4	Relief/2
CS-V-250	CVCS	Self	2	Relief/2
CS-V-426	CVCS	Motor	2	Globe/2
CS-V-437	CVCS	Manual	4	Weir/3
CS-V-439	CVCS	Manual	4	Weir/3
CS-V-440	CVCS	ΔP	4	Check/3
CS-V-1207	CVCS	Manual	4	Weir/3
CBS-V-58	CVCS	ΔP	8	Check/2
CBS-V-60	CVCS	ΔP	8	Check/2
CS-LCV—112B	CVCS	Motor	4	Gate/2
CS-LCV—112C	CVCS	Motor	4	Gate/2
CS-LCV—112D	CVCS	Motor	8	Gate/2
CS-LCV—112E	CVCS	Motor	8	Gate/2
CO-V-434	EFW	ΔP	4	Check/3
CO-V-435	EFW	ΔP	4	Check/3

Valve Number	System	Actuated by	Size(in)	Type/ANS Safety Class
IA-V-530	IA	Air	2	Globe/2
RC-V-22 (2)	RHRS	Motor	12	Gate/1
RC-V-23 (2)	RHRS	Motor	12	Gate/1
RC-V-87 (2)	RHRS	Motor	12	Gate/1
RC-V-88 (2)	RHRS	Motor	12	Gate/1
RH-FCV-618 (2)	RHRS	Air	8	Butterfly/2
RH-FCV-619 (2)	RHRS	Air	8	Butterfly/2
RH-HCV-606 (2)	RHRS	Air	8	Butterfly/2
RH-HCV-607(2)	RHRS	Air	8	Butterfly/2
RH-V-4	RHRS	ΔP	10	Check/2
RH-V-21	RHRS	Motor	8	Gate/2
RH-V-22	RHRS	Motor	8	Gate/2
RH-V-27	RHRS	Air	3/4	Globe/2
RH-V-28	RHRS	Air	3/4	Globe/2
RH-V-40	RHRS	ΔP	10	Check/2
RH-V-49	SIS	Air	3/4	Globe/2
CBS-V-2	SIS	Motor	12	Gate/2
CBS-V-5	SIS	Motor	12	Gate/2
CBS-V-47	SIS	Motor	8	Gate/2
CBS-V-48	SIS	ΔP	8	Check/2
CBS-V-49	SIS	Motor	6	Gate/2
CBS-V-51	SIS	Motor	8	Gate/2
CBS-V-52	SIS	ΔP	8	Check/2
CBS-V-53	SIS	Motor	6	Gate/2
CBS-V-55	SIS	ΔP	12	Check/2

(2) This valve is repositioned during a plant cooldown beyond hot standby and is required to remain in its normal position during Modes 1, 2, and 3. The normal passive function must be included in the IST Program. Its active function may be periodically tested by another Station procedure.

Valve Number	System	Actuated by	Size(in)	Type/ANS Safety Class
CBS-V-62	SIS	Self	3/4	Relief/2
CBS-V-145	SIS	ΔP	12	Check/2
CBS-V-56	SIS	ΔP	12	Check/2
CBS-V-146	SIS	ΔP	6	Check/2
CS-V-460	SIS	Motor	6	Gate/2
CS-V-461	SIS	Motor	6	Gate/2
CS-V-475	SIS	Motor	6	Gate/2
NG-V-13	SIS	Air	1	Globe/2
RH-V-13	SIS	Self	3/4	Relief/2
RH-V-14	SIS	Motor	8	Gate/2
RH-V-15	SIS	ΔP	6	Check/1
RH-V-25	SIS	Self	3/4	Relief/2
RH-V-26	SIS	Motor	8	Gate/2
RH-V-29	SIS	ΔP	6	Check/1
RH-V-30	SIS	ΔP	6	Check/1
RH-V-31	SIS	ΔP	6	Check/1
RH-V-32	SIS	Motor	8	Gate/2
RH-V-35	SIS	Motor	8	Gate/2
RH-V-36	SIS	Motor	8	Gate/2
RH-V-50	SIS	ΔP	8	Check/1
RH-V-51	SIS	ΔP	8	Check/1
RH-V-52	SIS	ΔP	6	Check/1
RH-V-53	SIS	ΔP	6	Check/1
RH-V-70	SIS	Motor	8	Gate/2
SI-V-3 (2)	SIS	Motor	10	Gate/1
SI-V-5	SIS	ΔP	10	Check/1

(2) This valve is repositioned during a plant cooldown beyond hot standby and is required to remain in its normal position during Modes 1, 2, and 3. The normal passive function must be included in the IST Program. Its active function may be periodically tested by another Station procedure.

Valve Number	System	Actuated by	Size(in)	Type/ANS Safety Class
SI-V-6	SIS	ΔP	10	Check/1
SI-V-10	SIS	Self	1	Relief/2
SI-V-17 (2)	SIS	Motor	10	Gate/1
SI-V-20	SIS	ΔP	10	Check/1
SI-V-21	SIS	ΔP	10	Check/1
SI-V-30	SIS	Self	1	Relief/2
SI-V-32 (2)	SIS	Motor	10	Gate/1
SI-V-35	SIS	ΔP	10	Check/1
SI-V-36	SIS	ΔP	12	Check/2
SI-V-45	SIS	Self	1	Relief/2
SI-V-47 (2)	SIS	Motor	10	Gate/1
SI-V-50	SIS	ΔP	10	Check/1
SI-V-51	SIS	ΔP	10	Check/2
SI-V-60	SIS	Self	1	Relief/2
SI-V-62	SIS	Air	3/4	Globe/2
SI-V-70	SIS	Air	3/4	Globe/2
SI-V-71	SIS	ΔP	4	Check/2
SI-V-76	SIS	Self	3/4	Relief/2
SI-V-77	SIS	Motor	4	Gate/2
SI-V-82	SIS	ΔP	6	Check/1
SI-V-87	SIS	ΔP	6	Check/1
SI-V-89	SIS	Motor	1½	Globe/2
SI-V-90	SIS	Motor	1½	Globe/2
SI-V-93	SIS	Motor	2	Globe/2
SI-V-96	SIS	ΔP	4	Check/2

(2) This valve is repositioned during a plant cooldown beyond hot standby and is required to remain in its normal position during Modes 1, 2, and 3. The normal passive function must be included in the IST Program. Its active function may be periodically tested by another Station procedure.

Valve Number	System	Actuated by	Size(in)	Type/ANS Safety Class
SI-V-101	SIS	Self	3/4	Relief/2
SI-V-102	SIS	Motor	4	Gate/2
SI-V-111	SIS	Motor	4	Gate/2
SI-V-112	SIS	Motor	4	Gate/2
SI-V-113	SIS	Self	3/4	Relief/2
SI-V-114	SIS	Motor	4	Gate/2
SI-V-131	SIS	Air	3/4	Globe/2
SI-V-134	SIS	Air	3/4	Globe/2
SI-V-138	SIS	Motor	4	Gate/2
SI-V-139	SIS	Motor	4	Gate/2
SI-V-140	SIS	ΔP	3	Check/1
SI-V-157	SIS	Air	3/4	Globe/2
SI-V-158	SIS	Air	3/4	Globe/2
SI-V-160	SIS	Air	3/4	Globe/2
SI-V-248	SIS	Self	3/4	Relief/2

TABLE 3.9(N)-12 MAXIMUM DEFLECTIONS ALLOWED FOR REACTOR INTERNAL SUPPORT STRUCTURES

Component	Allowable Deflections (in)	No-Loss-of Function Deflections (in)
Upper Barrel		
Radial inward	4.1	8.2
Radial outward	1.0	1.0
Upper package	0.10	0.15
Rod Cluster guide tubes	1.00	1.75

TABLE 3.10B-1 SEISMIC QUALIFICATION METHOD SUMMARY OF INSTRUMENTATION AND ELECTRICAL EQUIPMENT

CONTRACT NO.	EQUIPMENT DESCRIPTION	VENDOR	QUALIFICATION METHOD	REMARKS
106-2	Inverters - IE	Elgar	Analysis and Testing	Wyle Lab
109-1	Cable Trays	Metal Products	Static Test and Analysis	Metal Products
118-1	Penetrations	Westinghouse	Random Frequency Test and Analysis	Acton Lab & W
118-3	Electrical Conductor Seal Assy.	Conax	Random Frequency Test	Southwest Research Inst.
118-4	Electrical Connector	Conax	Random Frequency Test	Southwest Research Inst.
119-3	Emergency Power Sequencer	Brown Boveri	Random Frequency Test	Wyle Lab
119-5	DC Switchboards	Brown Boveri	Random Frequency Test	Wyle Lab
120-1	Power Distribution Panels	Gould	Random Frequency Test	Wyle Lab
120-9	R.C.P. Fuse Cabinets	Powell Electric	Random Frequency Test	Southwest Research Inst.
120-11	Isolation Relay Cabinets	Consolidated Controls	Testing	Farwell & Hendricks Inc.
129-1	Misc. Control Panels	Systems Control	Random Frequency Test	
137-1	Batteries	Gould	Random Frequency Test and Analysis	Wyle Lab and Gould
137-2	Battery Charges	Power Conversion Products	Random Frequency Test	Wyle Lab
143-1	Motor Control Centers	Brown Boveri	Random Frequency Test	Wyle Lab
144-1	5 kV and 15 Nonseg. Phase Bus	Brown Boveri	Analysis and Test	Wyle Lab and Gould
145-2	5 kV Switchgear	Brown Boveri	Random Frequency Test	Wyle Lab
145-3	480V Unit Substations	Brown Boveri	Random Frequency Test	Wyle Lab

CONTRACT NO.	EQUIPMENT DESCRIPTION	VENDOR	QUALIFICATION METHOD	REMARKS
162-1	Heat Tracing	Therman Manufacturing	Random Frequency Test	Southwest Research Inst.
170-1	Main Control Board	York Electro Panel	Dynamic Analysis and Testing	Analytical Eng. And Wyle Lab
170-4	Panel Mounted Small Case Recorders	Foxboro	Random Frequency Test	Southwest Research
170-5	Panel Mounted - Indicators	Sigma International Instruments	Random Frequency Test	Acton Lab
170-6	Miscellaneous - I&C Panels	Comsip	Dynamic Analysis and Testing	NQS, Analytical Eng. And Acton Lab
170-13	Intermediate Range - Nuclear Instr. Class 1E	Gamma-Metrics	Random Frequency Test Wyle Lab	
171-1	Instrument Racks	Mercury	Dynamic Analysis	NQS
172-1	Radiation Data Management System	General Atomic	Random Frequency Test	Wyle Lab
173-1	Control Valves	Control Components	Static Analysis	
173-4	Solenoid Valves	ASCO	Size-bent Test	
173-5	Control Valves - Nuclear Class 1, 2, 3	Masoneilan	Static Analysis	
173-7	Solenoid Valves	Valcor Engineering	Random Frequency Test	Dayton T. Brown
174-1	Electronic Transmitters	Foxboro	Random Frequency Test	Wyle Lab
174-2	Electronic Controllers & Accessories	Westinghouse	Random Frequency Test	Westinghouse
174-8	MSIV Logic Cabinets	Consolidated Controls	Random Frequency Test	Farwell & Hendricks
174-13	Electronic Transmitters	Rosemount	Random Frequency Test	Wyle Lab

CONTRACT NO.	EQUIPMENT DESCRIPTION	VENDOR	QUALIFICATION METHOD	REMARKS
174-14	HELB Detection Isolation System	Weed Instruments	Analysis and Test	
174-15	Level Transmitters Class 1E	Transamerica-DeLaval	Random Frequency Test	Wyle Lab
252-3	Flow Nozzles and Tubes	B.I.F.	Static Analysis	BIF
252-6	Rotometers	Ametek/Shutte-Koerting	Static analysis	NUS
252-8	Thermal Elements & Wells	Dravo/Thermo Electric	Static Analysis	Algor and Acton Lab
252-16	Differential Pressure Switches	ITT Barton	Random Frequency Test	Approved Engineering Test Labs
252-19 2331692	Seismic Monitoring System	Kinemetrics	Random Frequency Test	Wyle Lab
02209431	Loose Parts Monitoring System	Physical Acoustic Corporation (PAC) / MISTRAS	Analysis and Test	National Technical Systems
252-38	Temperature Switches(Class 1E)	ASCO	Sine Beat Test	
501-1	Class 1E Hydrogen Analyzers	Comsip	Static Analysis and Test	Engineering Analysis and Test

TABLE 3.10(N)-1 SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY

Equipment	EQDP*
Safety-Related Valve Electric Motor Operators	EQDP-HE-1 and 4
Safety-Related Pilot Solenoid Valves	EQDP-HE-2 and 5
Safety-Related Externally Mounted Limit Switches	EQDP-HE-3 and 6
Garrett (PORV) Solenoid-Operated Pilot Valve	EQDP-HE-9
Large Pump Motors (Outside Containment)	EQDP-AE-2
Canned Pump Motors (Outside Containment)	EQDP-AE-3
Electric Hydrogen Recombiner	EQDP-SP-1
Pressure Transmitters	EQDP-ESE-1 and 2
Differential Pressure Transmitters (Narrow Range)	EQDP-ESE-3 and 4
Resistance Temperature Detectors	EQDP-ESE-5 and 6
Excore Neutron Detectors (Power Range)	EQDP-ESE-8
Nuclear Instrumentation System (NIS) Cabinet	EQDP-ESE-10
Process Protection Sets	EQDP-ESE-13
Solid-State Protection System and Safeguard Test Cabinet (2 Train)	EQDP-ESE-16
Instrument Bus Power Supply (Static Inverter)	EQDP-ESE-18
Reactor Trip Switchgear	EQDP-ESE-20
Pressure Sensor - Containment (Narrow Range)	EQDP-ESE-21
Pressure Sensor - Containment (Wide Range)	EQDP-ESE-21
Differential Pressure Transmitters (Wide Range)	EQDP-ESE-4
*Equipment Qualification Data Package	