3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

3.1.1 Summary Description

This section contains an evaluation of the design bases of the South Texas Project Electric Generating Station (STPEGS) as measured against the NRC General Design Criteria (GDC) for Nuclear Power Plants, Appendix A of 10CFR50. There are 64 criteria in the GDC divided into six groups and intended to establish minimum requirements for the design of nuclear power plants.

It should be noted that the GDC were not written specifically for the pressurized water reactor; rather, they were intended to guide the design of all water-cooled nuclear power plants. As a result, the criteria are generic in nature and subject to a variety of interpretations. For this reason, there are some cases where conformance to a particular criterion is not directly measurable. In these cases, the conformance of plant design to the interpretation of the criterion is discussed. For each of the 64 criteria, a specific assessment of the plant design is made and a complete list of references is included to identify where detailed design information pertinent to each criterion is treated.

Based on the content herein, it is concluded that the STPEGS nuclear power plant fully satisfies and is in compliance with the GDC.

3.1.2 Criterion Conformance

3.1.2.1 Group I – Overall Requirements (Criteria 1-5)

3.1.2.1.1 Criterion 1 – Quality Standards and Records: Structures, systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A Quality Assurance (QA) Program shall be established and implemented in order to provide adequate assurance that these structures, systems and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

3.1.2.1.1.1 Evaluation Against Criterion 1 – Safety-related structures, systems and components are listed in Section 3.2. The QA Program for construction is described in the Quality Assurance Program Description (QAPD) and the QA Program for Operations is described in the Operations Quality Assurance Plan. Quality Assurance requirements have been applied to the safety-related items contained in the tables in Section 3.2. The intent of the QA Program is to assure sound engineering in all phases of design and construction through conformity to regulatory requirements and design bases described in the license application. In addition, the program assures adherence to specified standards of workmanship and implementation of recognized codes and standards in fabrication and construction. Such codes and standards have been evaluated to assure their applicability, adequacy, and sufficiency in keeping with the required safety function. It also includes the observance of proper pre-operational and operational testing and maintenance procedures as well as the documentation of the foregoing by keeping appropriate records. The total QA Program of the applicant and its principal contractors is responsive to and satisfies the quality-related requirements of 10CFR50, including Appendix B.

Section 3.2 contains a list of structures, systems and components that are classified with respect to their relationship to the safety function to be performed. Recognized codes and standards are applied to the equipment in these classifications as necessary to assure a quality product in keeping with the required safety function.

Documents are maintained which demonstrate that all the requirements of the QA Program are being satisfied. This documentation shows that appropriate codes, standards and regulatory requirements are observed, that specified materials and correct procedures are used, that qualified personnel are provided, and that the finished parts and components meet the applicable specifications for safe and reliable operation. These records are available so that any desired item of information is retrievable for reference. These records will be maintained during the life of the operating licenses.

The detailed QA Program developed by HL&P (historical context) and its contractors satisfies the requirements of Criterion 1. HL&P (historical context) has conducted audits of its principal contractors, Westinghouse Electric Corporation and Bechtel Energy Corporation, to establish the adequacy of their QA programs and to ensure that the programs are being implemented.

For further discussion, see the following sections:

System Quality Group Classifications	3.2.A.2 and 3.2.B.2
Seismic Design	3.7
Review and Audit	13.4
Initial Plant Test Program	14.2
Quality Assurance Program	17.2

3.1.2.1.2 Criterion 2 – Design Bases for Protection Against Natural Phenomena: Structures, systems and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems and components shall reflect: (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.

3.1.2.1.2.1 Evaluation Against Criterion 2 – The safety-related structures, systems and components are protected from or designed to either withstand the effects of natural phenomena without loss of capability to perform their safety functions or to fail in a safe condition. Natural phenomena taken into account in the design of these plant structures, systems, and components that are safety-related were determined from recorded data for the site vicinity with appropriate margin to account for uncertainties in historical data or were determined from guidance provided in applicable Regulatory Guides (RGs) such as 1.76, "Design Basis Tornado for Nuclear Power Plants".

The most severe natural phenomena postulated to occur at the site in terms of induced stresses are the Safe Shutdown Earthquake (SSE) and the Design Basis Tornado. Those structures, systems, and components essential for the mitigation and control of postulated accident conditions and those essential to maintain the integrity of the reactor coolant pressure boundary are designed to withstand the effects of the SSE. Those systems, structures, and components essential for the mitigation and control of the effects of postulated Loss-of-Coolant Accidents (LOCAs) are designed to withstand the effects of the LOCA as well as the effects associated with the SSE. These structures, systems, and components which perform a safety function are designed to withstand the effects of the most severe natural phenomena, including floods, hurricanes, tornadoes, and the SSE, as appropriate.

For further discussion, see the following sections of the UFSAR:

Meteorology	2.3
Hydrology	2.4
Geology and Seismology	2.5
Design of Structures, Components, Equipment and Systems	3.2 through 3.11

3.1.2.1.3 Criterion 3 – Fire Protection: Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the Containment and control room. Fire detection and fire fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems and components.

3.1.2.1.3.1 Evaluation Against Criterion 3 – The plant has been designed in accordance with the recommendations of the National Fire Protection Association and has been approved by the Association of Nuclear Insurers and the appropriate regulatory agencies having jurisdiction.

Noncombustible and fire-resistant materials have been used wherever practical throughout the facility, particularly in areas containing critical portions of the plant such as the Containment structure, control room and components of safety-related systems. These systems are designed and located to minimize the effects of fires or explosions on their redundant components. Facilities for the storage of combustible materials such as fuel oil are located, designed and protected to minimize both the probability and the effects of a fire.

Equipment and facilities for fire detection, alarm and extinguishment are provided to protect both plant and personnel from fire or explosion. Fire Protection and Detection System reliability is ensured by periodic tests and inspections.

Administrative controls are used where applicable throughout the facility to minimize the probability and consequences of fires or explosions.

The Fire Protection System is designed such that a failure of any component of the system:

- Will not cause an accident resulting in significant release of radioactivity to the environment.

- Will not impair the ability of redundant equipment to safely shut down the reactor or limit the release of radioactivity to the environment in the event of a LOCA.

For further discussion, see the following sections of the UFSAR.

Materials, Quality Control and Special Construction Techniques	3.8.1.6
Independence of Redundant Safety-Related Systems	7.1.2.2
Separation of Redundant Systems	8.3.1.4
Fire Protection System	9.5.1

3.1.2.1.4 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 LOCAs. These structures, systems and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, the dynamic effects associated with postulated pipe ruptures of primary coolant loop piping in pressurized water reactors may be excluded from the design basis when analyses demonstrate the probability of rupturing such piping is extremely low under design basis conditions.

3.1.2.1.4.1 Evaluation Against Criterion 4 – Safety-related structures, systems, and components are designed to accommodate the effects of and to be compatible with the environmental conditions (including the pressure, temperature, humidity and radiation conditions) associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. Protection criteria are presented in Sections 3.5 and 3.6 and the environmental conditions are described in Section 3.11.

These structures, systems, and components are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit. The dynamic effects associated with postulated ruptures in the RCS main loop piping are shown to be of extremely low probability of occurring under design conditions and are not included in the design basis. Details of the design, environmental testing, and construction of these systems, structures and components are included in other sections of the UFSAR:

Water Level (Flood) Design	3.4
Missile Protection Criteria	3.5
Criteria for Protection Against Dynamic Effects Associated with Postulated Rupture of Piping	3.6
Design of Category I Structures	3.8
Mechanical Systems and Components	3.9
Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment	3.10
Environmental Design of Mechanical and Electrical Equipment	3.11
Independence of Redundant Safety-Related Systems	7.1.2.2
Separation of Redundant Systems	8.3.1.4
Accident Analysis	15.0

3.1.2.1.5 Criterion 5 – Sharing of Structures, Systems or Components: Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

3.1.2.1.5.1 Evaluation Against Criterion 5 – The ultimate heat sink is the only shared safety-related system.

For further discussion, see Section 9.2.5.

3.1.2.2 Group II – Protection by Multiple Fission Product Barriers (Criteria 10-19).

3.1.2.2.1 Criterion 10 – Reactor Design: The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margins to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

3.1.2.2.1.1 Evaluation Against Criterion 10 – The reactor core and associated coolant, control, and protection systems are designed with adequate margins to:

1. Preclude significant fuel damage during normal core operation and operational transients (Condition I)* or any transient conditions arising from occurrences of moderate frequency (Condition II)*.

2. 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 immediate resumption of operation.

3. 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 reactor components including the fuel and reactivity control materials. Section 3.9 discusses the design bases and design evaluation of the reactor vessel internals and the control rod drive mechanisms. 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.

3.1.2.2.2 Criterion 11 – Reactor Inherent Protection: The reactor core and associated coolant systems shall be designed so that in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

3.1.2.2.2.1 Evaluation Against Criterion 11 – Prompt compensatory reactivity feedback effects are assured when the reactor is critical by the negative fuel temperature effect (Doppler effect) and by the nonpositive operational limit on moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is assured by the inherent design using low-enrichment fuel; the nonpositive moderator temperature coefficient of reactivity is assured by administratively controlling the dissolved absorber concentration or by burnable poisons.

These reactivity coefficients are discussed in Section 4.3.

3.1.2.2.3 Criterion 12 – Suppression of Reactor Power Oscillations: The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

3.1.2.2.3.1 Evaluation Against Criterion 12 – Power oscillations of the fundamental mode are inherently eliminated by the negative Doppler and non-positive moderator temperature coefficient of reactivity.

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* Defined by ANSI N18.2 – 1973

** Defined by ANSI N18.2-1973 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 the radial, diametral, and azimuthal overtone modes are heavily damped due to the inherent design and due to the negative Doppler and nonpositive moderator temperature coefficients of reactivity.

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 discussed in Section 4.3.

Control rods provide the capability of attenuating axial oscillations.

3.1.2.2.4 Criterion 13 – Instrumentation and Control: Instrumentation and control shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary (RCPB), and the Containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

3.1.2.2.4.1 Evaluation Against Criterion 13 – 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 for the Reactor Coolant System (RCS), Steam and Power Conversion System, the Containment, Engineered Safety Features (ESF) Systems, Radioactive 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 with the controls for maintaining the indicated parameter in the proper range.

The quantity and types of process instrumentation provided ensure safe and orderly operation of all systems over the full design range of the plant. These systems are described in Chapters 5, 6, 7, 8, 9, 10, 11, and 12.

3.1.2.2.5 Criterion 14 – Reactor Coolant Pressure Boundary: The RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture.

3.1.2.2.5.1 Evaluation Against Criterion 14 – The RCPB 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 Sections 3.9 and 5.2 for details.

In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions, such as pipe rupture and seismic events, 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 (refer to Section 5.2.2). The RCPB has provisions for inspection, testing and surveillance of critical areas to assess the structural and leaktight 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.

3.1.2.2.6 Criterion 15 – Reactor Coolant System Design: The RCS and associated auxiliary, control and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.

3.1.2.2.6.1 Evaluation Against Criterion 15 – 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, RCPB components achieve a large margin of safety by the use of proven American Society of Mechanical Engineers (ASME) materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and integrated hydrostatic testing of assembled components. Chapter 5 discusses the Reactor Coolant System design.

3.1.2.2.7 Criterion 16 – Containment Design: Containment and associated systems shall be provided to establish and essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the Containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

3.1.2.2.7.1 Evaluation Against Criterion 16 – The Containment Isolation System will limit leakage to the small percentages given in Section 6.2.4 by providing an essentially leaktight barrier against radioactivity which may be released to the Containment atmosphere in the unlikely event of an accident.

Additional systems provided to prevent the uncontrolled release of radioactivity from the Containment to the environment are the Emergency Core Cooling System (ECCS), Containment Spray System (CSS), and Containment Heat Removal System (CHRS). These systems mitigate the potential consequences of a LOCA or main steam line break. The Containment and these associated engineered safeguard systems are designed to operate under all internal and external environmental conditions that may be postulated to occur during the life of the plant, including both short-and long-term effects following a LOCA. Containment leak rate testing is performed in accordance with 10CFR50, Appendix J.

For further discussion, see the following sections of the UFSAR:

Concrete Containment	3.8.1
Concrete and Structural Steel Internal Structures of Concrete Containment	3.8.3
Containment Systems	6.2
Accident Analyses	15.0

3.1.2.2.8 Criterion 17 – Electric Power Systems: An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the RCPB are not exceeded as a result of anticipated operational occurrences, and (2) the core is cooled and Containment integrity and other vital functions are maintained in the event of postulated accidents.

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

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

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

3.1.2.2.8.1 Evaluation Against Criterion 17 – Onsite and offsite electrical power systems are provided, and each is designed with adequate independence, capacity, redundancy, and testability to assure the functioning of safety-related systems.

Offsite power is transmitted to the plant switchyard at 345 kV by multiple circuits on four separate rights-of-way. The two unit standby transformers are energized from separate buses in the switchyard via independent feeders. Each standby transformer has the capacity to supply the Class 1E loads of both units. In normal operation, the Class 1E loads of each unit can be supplied by the standby transformers and/or its auxiliary unit transformer. In the event of a loss of power from its normal source that unit’s Class 1E loads are manually transferred to that unit’s auxiliary transformer or to the standby transformers.

In the event of a loss of offsite power, three standby diesel generator sets are provided for each unit. Any two diesel generators will provide sufficient power to a unit for safe shutdown or, in the event of an accident, to mitigate the consequences to within acceptable limits. Four ESF batteries are provided for each unit to supply Class 1E dc power. There are no interconnections between units of the standby emergency power systems.

The Standby AC and DC Power Systems consist of independent and redundant power sources and distribution equipment such that no single failure prevents the systems from performing their safety functions.

For further details see Sections 8.1, 8.2, and 8.3.

The power systems, as designed, conform to Criterion 17.

3.1.2.2.9 Criterion 18 – Inspection and Testing of Electric Power Systems: Electric power systems important to safety shall be designed to permit appropriate periodic inspections and testing of important areas and features such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

3.1.2.2.9.1 Evaluation Against Criterion 18 – The ESF power supply buses and associated diesel generators are arranged for periodic testing of each system independently. The testing procedure simulates a loss of bus voltage to start the diesel, bring it to operating condition and automatically connect it to the bus. Full-load testing of the diesel generator can be performed by manually synchronizing to the normal supply. These tests, performed periodically in accordance with the Technical Specifications, will prove the operability of the power supply system under conditions as close to design as practical to assess the continuity of the system and condition of the components.

The design of the emergency power systems provides testability in accordance with the requirements of Criterion 18. For further discussion, see Section 8.3.

For further discussion, see the following sections of the UFSAR:

AC Power Systems	8.3.1
DC Power Systems	8.3.2
Initial Test Program	14.0

3.1.2.2.10 Criterion 19 – Control Room: A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in safe condition under accident conditions, including LOCA. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

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

3.1.2.2.10.1 Evaluation Against Criterion 19 – The control room contains the following equipment: control panels which contain those instruments and controls necessary for operation and surveillance of the plant functions, such as the reactor and its auxiliary systems (ESF, Turbine-Generator, Steam and Power Conversion Systems), and station electrical distribution boards.

The design of the control room permits safe occupancy during abnormal conditions. The Control Room Ventilation System is designed to recirculate control room air and filter make-up air through high-efficiency particulate air filters and charcoal adsorbers when required. Radiation detectors, alarms and emergency lighting are provided. Alternate local controls and local instruments are available for equipment required to bring the plant to and maintain a hot standby condition. It is also possible to attain a cold shutdown condition from locations outside the control room through the use of suitable procedures. Control room shielding and the Ventilation System are designed to maintain tolerable radiation exposure levels (maximum of 5 rem whole body or its equivalent to any part of the body) for the duration of the accident.

For further discussion, see the following sections of the UFSAR:

Control Room Habitability Systems	6.4
Instrumentation and Control	7.0
Systems Required for Safe Shutdown	7.4
Control Room HVAC System	9.4.1
Fire Protection System	9.5.1
Plant Lighting Systems	9.5.3
Shielding	12.3.2
Accident Analysis	15.0

3.1.2.3 Group III – Protection and Reactivity Control Systems (Criteria 20-29).

3.1.2.3.1 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.

3.1.2.3.1.1 Evaluation against Criterion 20 – 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 Institute of Electrical and Electronic Engineers (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 exceeds 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 rod cluster control assemblies. This causes the rods to insert by gravity, which rapidly reduces the reactor power output. The response and adequacy of the Protection System has been verified by analysis of anticipated transients.

The ESF Actuation System automatically initiates emergency core-cooling and other safeguard 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 trips the reactor on manual or automatic safety injection signal generation.

3.1.2.3.2 Criterion 21 – Protection System Reliability and Testability: The Protection System shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundance 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.

3.1.2.3.2.1 Evaluation Against Criterion 21 – The Protection System is designed for high functional reliability and inservice testability such that the requirements of Criterion 21 are satisfied.

Compliance with this criterion is discussed in detail in Section 7.2.2.2.1, 7.2.2.2.3 and 7.3.2.2.

3.1.2.3.3 Criterion 22 – Protection System Independence: The Protection System shall be designed to assure that the effects of natural phenomena and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

3.1.2.3.3.1 Evaluation Against Criterion 22 – The 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 was designed into the system. The extent of this functional diversity was evaluated for a wide variety of postulated accidents. Diverse protection functions will automatically terminate an accident before intolerable consequences can occur.

Automatic reactor trips are based upon neutron flux measurements, reactor coolant loop temperature measurements, pressurizer pressure and level measurements, and reactor coolant pump power underfrequency and under-voltage measurements. Trips may also be initiated manually or by safety injection signal. See Section 7.2 for details of the Reactor Trip System and Section 7.3 for details of the Engineered Safety Features Actuation System.

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 if required. Typical protection system equipment is subjected to type tests under simulated seismic conditions using conservatively large accelerations and applicable frequencies. The test results indicate no loss of the protection function. Refer to Sections 3.10 and 3.11 for further details.

3.1.2.3.4 Criterion 23 – Protection System Failure Modes: The Protection System shall be designed to fall into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air) or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

3.1.2.3.4.1 Evaluation Against Criterion 23 – The 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 deenergize-to-trip principle so 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.

3.1.2.3.5 Criterion 24 – Separation of Protection and Control Systems: The Protection System shall be separated from control systems to the extent that failure of any single Control System component or channel, or failure or removal from service of any single Protection System component or channel which is common to the Control and Protection System 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.

3.1.2.3.5.1 Evaluation Against Criterion 24 – 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 devices which are classified as protection components. The adequacy of system isolation was 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. Distinction between channel and train is made in this discussion. The removal of a train from service is addressed by the Technical Specifications.

3.1.2.3.6 Criterion 25 – Protection System Requirements for Reactivity Control Malfunctions: The Protection System shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the Reactivity Control Systems, such as accidental withdrawal (not ejection or dropout) of control rods.

3.1.2.3.6.1 Evaluation Against Criterion 25 – The Protection System is designed to limit reactivity transients so that fuel design limits are not exceeded. Reactor shutdown by 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 be generated independently. 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.

3.1.2.3.7 Criterion 26 – Reactivity Control System Redundancy and Capability: Two independent Reactivity Control Systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

3.1.2.3.7.1 Evaluation Against Criterion 26 – Two 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 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 control banks are designed to shut down 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.

The Chemical and Volume Control System (CVCS) will 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 RCCSs 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.

3.1.2.3.8 Criterion 27 – combined Reactivity Control Systems Capability: The Reactivity Control Systems shall be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

3.1.2.3.8.1 Evaluation Against Criterion 27 – The facility is provided with means for 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, 6, 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.

3.1.2.3.9 Criterion 28 – Reactivity Limits: The Reactivity Control Systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the RCPB greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

3.1.2.3.9.1 Evaluation Against Criterion 28 – 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 RCPB 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 RCCAs and the dilution of the boric acid in the RCS are limited by the physical design characteristics of the RCCAs and of the CVCS. 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 CVCS to avoid an inadvertent excessive rate of boron dilution is discussed in Chapter 15.

Assurance of adequate core-cooling capability following Condition IV accidents, such as rod ejections, steam line break, etc., is provided through analysis to demonstrate that the RCPB stresses remain 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. Condition IV accidents are discussed in Section 15.0.1.4.

3.1.2.3.10 Criterion 29 – Protection Against Anticipated Operational Occurrences: The Protection and Reactivity Control Systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

3.1.2.3.10.1 Evaluation Against Criterion 29 – The Protection and Reactivity Control Systems are designed to assure extremely high probability of performing their required safety functions in any anticipated operational occurrences. 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.

3.1.2.4 Group IV – Fluid Systems (Criteria 30-46).

3.1.2.4.1 Criterion 30 – Quality of Reactor Coolant Pressure Boundary: Components which are part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

3.1.2.4.1.1 Evaluation Against Criterion 30 – By using conservative design practices and detailed quality control procedures, the pressure-retaining components of the RCPB are designed and fabricated to retain their integrity during normal and postulated accident conditions. Components for the RCPB are designed, fabricated, inspected and tested in conformance with ASME B&PV Code, Section III. All components are classified according to ANSI N18.2-1973 and ANSI N18.2a-1975 and are accorded the quality measures appropriate to the classification. The design bases and evaluations of RCPB components are discussed in Chapter 5. Further, product and process quality planning is provided as described in the Operations Quality Assurance Plan to assure conformance with the applicable codes and standards, and to retain appropriate documented evidence verifying compliance. Further discussion on this subject is provided in the response to Criterion 14, "Reactor Coolant Pressure Boundary."

Means are provided for detecting reactor coolant leakage. The Leak Detection System consists of sensors and instruments to detect and annunciate potentially hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, by increased frequency of sump pump operation, and by measuring fission product concentration. In addition to these means of detection, large leaks are detected by changes in flowrates in process lines, and changes in pressurizer level. The allowable leak rates were based on the predicted and experimentally determined behavior of cracks in pipes, the ability to make up coolant system leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. While the Leak Detection System provides protection from small leaks, the ECCS network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges.

The RCPB and the Leak Detection System are designed to meet the requirements of Criterion 30.

For further discussion, see the following UFSAR sections:

Design of Structures, Components, Equipment, and Systems	3.0
Integrity of Reactor Coolant Pressure Boundary	5.2
RCPB Leakage Detection System	5.2.5
Reactor Vessel and Appurtenances	5.3
Reactor Coolant Piping	5.4.3
Pressurizer Water Level Control	7.7.1.6
Quality Assurance Program	17.2

3.1.2.4.2 Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary: The RCPB shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner, and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (a) material properties, (b) the effects of irradiation on material properties, (c) residual, steady-state and transient stresses, and (d) size of flaws.

3.1.2.4.2.1 Evaluation Against Criterion 31 – Close control is maintained over material selection and fabrication for the RCS to assure that the boundary behaves in a nonbrittle manner and precludes the rapid propogation of fractures (see Section 3.1.2.2.5.1). Materials for the RCS 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 10CFR50, Appendix G. Detection and determination of leak size is discussed in the evaluation against Criterion 30.

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 the straight beam code requirements, the performance of a 100-percent volumetric angle beam inspection of reactor vessel plate material and a post-hydrostatic test ultrasonic map of all full-penetration welds in the pressure vessel are required.

2. Radiation Surveillance Program – In the surveillance programs, the evaluation of the Reactor Pressure Vessel (RPV) radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch and tensile 1/2 T (thickness) impact tension fracture mechanics 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 the American Society for Testing and Materials E-185-73, “Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels,” and the requirements of 10CFR50, Appendix H.

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 RCS are equivalent to those used for the reactor vessel. The inspections of reactor vessel, pressurizer, piping, pumps, and steam generator are governed by ASME Section III requirements. See Chapter 5 for details.

Allowable pressure/temperature relationships for plant heatup and cooldown rates are calculated using methods presented in 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 (K_{IR}) 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.

3.1.2.4.3 Criterion 32 – Inspection of Reactor Coolant Pressure Boundary: Components which are part of the RCPB shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

3.1.2.4.3.1 Evaluation Against Criterion 32 – The design of the RCPB 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 nozzles to reactor coolant piping welds and certain portions of 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 RCPB components' integrity. The RCPB is periodically inspected under the provisions of ASME Section XI.

Monitoring of changes in the fracture toughness properties of the reactor vessel core region plates, weldments, and associated heat-affected zones are performed in accordance with 10CFR50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." Samples of reactor vessel plate materials are retained and catalogued in the event 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 the appropriate sections in Chapter 5 for further details on inspection and surveillance requirements.

3.1.2.4.4 Criterion 33 – Reactor Coolant Makeup: A system to supply reactor coolant makeup for protection against small breaks in the RCPB shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for Onsite Electric Power System operation (assuming offsite power is not available), and for Offsite Electric Power System operation (assuming onsite power is not available), the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

3.1.2.4.4.1 Evaluation Against Criterion 33 – The CVCS 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. 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 Section 9.3.4 with details of the Electric Power System included in Chapter 8.

3.1.2.4.5 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 RCPB are not exceeded.

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

3.1.2.4.5.1 Evaluation Against Criterion 34 – The Residual Heat Removal System (RHRS), in conjunction with the Steam and Power Conversion System, is designed to transfer the fission product decay heat and other residual heat from the reactor core and to maintain the RCS temperature within acceptable limits. The cross-over from the Steam and Power Conversion System to the RHRS occurs during the cooldown to hot shutdown conditions.

Suitable redundancy at temperatures below approximately 350°F is accomplished with the three residual heat removal pumps (located in separate compartments with means available for draining and monitoring leakage), the three heat exchangers and the associated piping, cabling, and electric power sources. The RHRS is able to operate on either the Onsite or Offsite Electrical Power System.

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

Details of the system designs are given in Section 5.4.7 and Chapter 10.

3.1.2.4.6 Criterion 35 – Emergency Core Cooling: A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (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.

3.1.2.4.6.1 Evaluation Against Criterion 35 – The ECCS is provided to cope with any LOCA 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 1 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 systems are included in Section 6.3. An evaluation of the adequacy of the system functions is included in Chapter 15. Performance evaluations have been conducted in accordance with 10CFR50.46 and 10CFR50 Appendix K.

3.1.2.4.7 Criterion 36 – inspection of Emergency Core Cooling System: The ECCS shall be designed to permit appropriate periodic inspection of important components, such as water injection nozzles, and piping, to assure integrity and capability of the system.

3.1.2.4.7.1 Evaluation Against Criterion 36 – Design provisions facilitate access to the critical parts of the injection nozzles, pipes, and valves for visual inspection and for nondestructive inspection where such techniques are desirable and appropriate. The design is in accordance with ASME Code, Section XI requirements.

The components outside the Containment are accessible for leaktightness inspection during operation of the reactor.

Details of the inspection program for the ECCS are discussed in Section 6.3.

3.1.2.4.8 Criterion 37 – Testing of Emergency Core Cooling System: The ECCS shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight 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.

3.1.2.4.8.1 Evaluation Against Criterion 37 – The ECCS is provided with sufficient test connections and isolation valves to permit appropriate periodic pressure testing to assure the structural and leaktight integrity of its components. In addition, the system is designed to permit periodic testing to assure the operability and performance of the active components of the system. The system is tested periodically to verify the performance of the full operational sequence that brings the system into operation using power supplied from the standby generators and the offsite power systems.

For further discussion, see the following sections of the UFSAR:

Performance Evaluation (ECCS)	6.3.3
Engineered Safety Feature Actuation System (ESFAS)	7.3
Onsite Power Systems	8.3

3.1.2.4.9 Criterion 38 – Containment Heat Removal: A system to remove heat from the Containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the Containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

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

3.1.2.4.9.1 Evaluation Against Criterion 38 – The CHRS consists of the CSS, the Reactor Containment Fan Cooler (RCFC) Subsystem and the residual heat removal (RHR) heat exchangers. The CHRS acts in conjunction with the Safety Injection System to remove heat from the Containment. The CHRS is designed to accomplish the following functions in the unlikely event of a LOCA: to rapidly condense the steam within the Containment in order to prevent over-pressurization during blowdown of the RCS; and to provide long-term continuous heat removal from the Containment.

Initially, the CSS and the high-and low-head safety injection (HHSI and LHSI) pumps take suction from the refueling water storage tank (RWST). During the recirculation phase, the CSS and the HHSI and LHSI pumps take suction from the Containment emergency sumps.

The CHRS is divided into three trains. Each train is sized to remove 50 percent of the system design heat load at the start of recirculation. Each train of the CHRS is supplied power from a separate independent Class 1E bus. The redundancy and capability of the Offsite and Emergency Power Systems are presented in the evaluation against Criterion 17. Redundant system trains and emergency diesel power supplies provide assurance that system safety functions can be accomplished.

For further discussion, see the following sections of the UFSAR:

Residual Heat Removal System	5.4.7
Containment Systems	6.2
Engineered Safety Features Actuation System	7.3
Onsite Power System	8.3
Accident Analysis	15.0

3.1.2.4.10 Criterion 39 – Inspection of Containment Heat Removal Systems: The CHRS shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

3.1.2.4.10.1 Evaluation Against Criterion 39 – Provisions are made to facilitate periodic inspections of active components and other important equipment in the CHRS. During plant operations, the pumps, valves, piping, instrumentation, wiring and other components outside the Containment can be visually inspected at any time and are inspected periodically. The functional testing of most components is correlated with component inspection.

The CHRS is designed to permit periodic inspection of major components.

For further discussion, see the following sections of the UFSAR:

Residual Heat Removal System	5.4.7
Containment Systems	6.2
Engineered Safety Feature Actuation System (ESFAS)	7.3

3.1.2.4.11 Criterion 40 – Testing of Containment Heat Removal Systems: The CHRS shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system and (3) the operability of the system as a whole, and, under conditions as close to the design as practicable, performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the Protection System, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

3.1.2.4.11.1 Evaluation Against Criterion 40 – The CHRS is provided with sufficient test connections and isolation valves to permit periodic pressure testing. System piping, valves, pumps, heat exchangers, and other components of the CHRS are arranged so that each component can be tested periodically for operability, including the transfer to the standby power system. The delivery capability of the CSS is tested periodically to the extent practicable up to the last isolation valves before the spray nozzles.

The delivery capability of the spray nozzles are tested after a maintenance activity to ensure the spray nozzles are unobstructed, by blowing low-pressure air through the nozzles and verifying the flow (Ref. 3.3-1 and 3.3-2). The CSS is tested for operational sequence as close to design condition as practicable. The RCFCs are tested for operation under full-load conditions during preoperational Containment leak rate testing. The operation of associated cooling water systems is discussed in response to Design Criterion 46. CN-3179

For further discussion, see the following sections of the UFSAR:

Containment Systems	6.2
Engineered Safety Features Actuation System	7.3
Onsite Power System	8.3

3.1.2.4.12 Criterion 41 – Containment Atmosphere Cleanup: Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the Containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the Containment atmosphere following postulated accidents to assure that Containment integrity is maintained.

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

3.1.2.4.12.1 Evaluation Against Criterion 41 – The CSS is provided to reduce the concentration and quantity of fission products in the Containment atmosphere following a LOCA. Per 10CFR50.44, hydrogen recombiners are no longer required for design basis accidents.

The equilibrium sump pH is maintained by trisodium phosphate (TSP) contained in baskets on the containment floor. The initial CSS water and spilled RCS water dissolves the TSP into the containment sump allowing recirculation of the alkaline fluid. Each unit is equipped with three 50-percent spray trains taking suction from the Containment sump. Each Containment spray train is supplied power from a separate bus. Each bus is connected to both the Offsite and the Standby Power Supply Systems. This assures that for Onsite or for Offsite Electrical Power System failure, their safety function can be accomplished, assuming a single failure.

Post-accident combustible gas control is assured by the use of the Supplementary Containment Purge Subsystem.

For further discussion, see the following sections of the UFSAR:

Containment Systems	6.2
Containment Spray System – Iodine Removal	6.5.2
Containment Hydrogen Sampling System	7.6.5
Containment HVAC System	9.4.5
Accident Analyses	15.0

3.1.2.4.13 Criterion 42 – Inspection of Containment Atmosphere Cleanup System: The Containment Atmosphere Cleanup Systems shall be designed to permit appropriate periodic inspection of important components, such as filters, frames, ducts, and piping to assure the integrity and capability of the systems.

3.1.2.4.13.1 Evaluation Against Criterion 42 – The Containment Atmosphere Cleanup Systems discussed under Criterion 41 are designed and located so that they can be inspected periodically. Ducts, plenums and casings are provided with access doors for internal inspection and with test connections to measure flow. The CSS's trisodium phosphate and baskets are inspected during refueling outages. The same section references apply as those given in response to Criterion 41.

3.1.2.4.14 Criterion 43 – Testing of Containment Atmosphere Cleanup Systems: The Containment Atmosphere Cleanup Systems shall be designed to permit appropriate periodic pressure and functional testing to assure: (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) 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.

3.1.2.4.14.1 Evaluation Against Criterion 43 – The operation of the CSS pumps can be tested by use of a recirculation line to the RWST. The system valves can be operated through their full travel. The system is checked for leaktightness during testing and with an air flow test described in Section 3.1.2.4.11.1.

The Hydrogen Monitoring System can be periodically tested for functional performance.

The design airflow and the full operational sequence that would bring the systems into action including the transfer to standby power sources can be tested.

For further discussion, see the following section of the UFSAR:

Containment Systems	6.2
Containment Spray System – Iodine Removal	6.5.2
Engineered Safety Features Actuation System	7.3

3.1.2.4.15 Criterion 44 – Cooling Water: A system to transfer heat from structures, systems and components important to safety to an ultimate heat sink (UHS) shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems and components under normal operating and accident conditions.

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

3.1.2.4.15.1 Evaluation Against Criterion 44 – The safety-related cooling water systems are the Essential Cooling Water System (ECWS) and the Component Cooling Water System (CCWS). The CCWS is a closed cooling water system and is designed to remove heat from equipment directly associated with Containment cooling following a Design Basis Accident (DBA). The ECWS is designed to remove heat from the CCWS and from other equipment that is required to operate following a DBA. The heat from the ECWS is rejected to the UHS. The UHS is designed to dissipate the rejected heat from the simultaneous shutdown and cooldown of both units or the shutdown and cooldown of one unit simultaneously with the dissipation of post-accident heat from the other unit. The ECWS and the CCWS are arranged in three redundant trains. Each train is capable of removing 50 percent of the total heat load following the design basis LOCA. The system is sized to transfer the design heat load from structures, systems and safety-related components to the UHS during normal shutdown and accident conditions assuming a single failure in one of the redundant loops. The portions of the essential cooling pond facility used as the UHS are designed to seismic Category I requirements.

Both systems are arranged so that all components in corresponding loops are supplied power by a separate power bus. The bus shall be capable of receiving power from both the offsite power and the standby diesel generator power supply.

For further discussion, see the following sections of the UFSAR:

Engineered Safety Features Actuation System	7.3
Onsite Power Systems	8.3
Water Systems	9.2

3.1.2.4.16 Criterion 45 – Inspection of Cooling Water System: The Cooling Water System shall be designed to permit periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

3.1.2.4.16.1 Evaluation Against Criterion 45 – The ECWS and the CCWS are designed to permit appropriate periodic inspection of important components, including pumps, strainers, heat exchangers and isolation valves to assure the integrity and capability of the systems. The UHS is an open reservoir within the site boundary and is thus accessible to inspection.

All important components are located in accessible locations to facilitate periodic inspection during normal plant operation. Suitable manholes, handholes, inspection ports, or other design and layout features are provided for this purpose.

For further discussion, see Section 9.2.

3.1.2.4.17 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 leaktight integrity of its components, (2) the operability and the performance of the active components of the system and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for LOCAs, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

3.1.2.4.17.1 Evaluation Against Criterion 46 – The ECWS is designed to permit periodic inspection and testing of all components to assure the structural and leaktight integrity and reliability of the system. Data is taken periodically during normal plant operation to confirm heat transfer capability. In addition to the tests and inspection of individual system components, periodic functional testing is performed to assure the operability of the system. The test assure, under conditions as close to operating as practical, the system's operability during the full operational sequence for normal reactor shutdown and postulated DBAs, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

For further discussion, see the following sections of the UFSAR:

Engineered Safety Features Actuation System	7.3
Onsite Power Systems	8.3
Water Systems	9.2

3.1.2.5 Group V – Containment (Criteria 50-57).

3.1.2.5.1 Criterion 50 – Containment Design Bases: The Reactor Containment structure, including access openings, penetrations and the CHRS, shall be designed so that the Containment structure and its internal compartments can accommodate, without exceeding the design leakage rate with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin shall reflect consideration of : (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators (SGs) and energy from metal/water and other chemical reactions that may result from degraded emergency core-cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and Containment responses, and (3) the conservatism of the calculation model and input parameters.

3.1.2.5.1.1 Evaluation Against Criterion 50 – The Containment structure, including access openings and penetrations, is designed to accommodate, without exceeding the design leak rate, the transient peak pressure and temperature associated with a DBA.

The Containment subcompartments (e.g., SG compartment, volume under the reactor vessel and the pressurizer compartment) are designed to withstand peak differential pressures resulting from the postulated hot or cold leg breaks and pressurizer line breaks with sufficient margin.

The Containment structure and ESF Systems were evaluated for various combinations of energy release. The analysis accounts for system thermal and chemical energy and for nuclear decay heat.

The maximum temperature and pressure reached in the Containment during the worst-case accident are shown in Section 6.2 to be well below the design temperature and pressure of this structure.

The cooling capacity of the CHRS is adequate to prevent overpressurization of the structure and to return the Containment to near atmospheric pressure within 1 day following the accident. For further discussion, see the following section of the UFSAR:

Concrete Containment	3.8.1
Containment Systems	6.2

3.1.2.5.2 Criterion 51 – Fracture Prevention of Containment Pressure Boundary: The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions, (1) its ferritic materials behave in a nonbrittle manner, and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state and transient stresses and (3) size of flaws.

3.1.2.5.2.1 Evaluation Against Criterion 51 – The Containment liner material has a nil-ductility transition (NDT) temperature at least 30°F below the minimum service temperature.

Principal Containment load-carrying components of ferritic materials exposed to the external environment were selected so that their temperatures under normal operating and testing conditions are not less than 30°F above NDT temperature.

For further discussion see the following sections of the UFSAR:

Concrete Containment	3.8.1
Quality Assurance	17.2

3.1.2.5.3 Criterion 52 – Capability for Containment Leakage Rate Testing: The Reactor Containment and other equipment which may be subjected to Containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at Containment design pressure.

3.1.2.5.3.1 Evaluation Against Criterion 52 – The Containment and its penetrations are designed and constructed, and the necessary equipment provided to permit periodic integrated leak rate tests during plant lifetime.

The testing program is conducted in accordance with 10CFR50, Appendix J.

Provisions are made in the Containment design to permit periodic leak rate tests at Containment design pressure to verify the continued leaktight integrity of the Containment.

For further discussion, see the following sections of the UFSAR:

Testing and Inspection	6.2.1.6

3.1.2.5.4 Criterion 53 – Provisions for Containment Testing and Inspection: The 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 leaktightness of penetrations which have resilient seals and expansion bellows.

3.1.2.5.4.1 Evaluation Against Criterion 53 – The design of the Containment permits the periodic inspection, surveillance and testing of the leaktightness of the Containment and its penetrations which have resilient seals and expansion bellows in accordance with the requirements of 10CFR50, Appendix J.

For further discussion, see the following section of the UFSAR:

Testing and Inspection	6.2.1.6

3.1.2.5.5 Criterion 54 – Piping Systems Penetrating Containment: Piping systems penetrating primary Reactor Containment shall be provided with leak detection, isolation and Containment capabilities having redundancy, reliability and performance capabilities 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.

3.1.2.5.5.1 Evaluation Against Criterion 54 – Piping systems penetrating the Containment are provided with Containment isolation barriers. These piping systems are designed to withstand a pressure at least equal to the maximum Containment pressure.

Those penetrations that must be closed for Containment isolation have redundant valving and/or associated apparatus as described in Section 6.2.4, so that no single active failure can result in either loss of isolation or excessive leakage. Each valve is tested periodically during normal operation or during shutdown to ensure its operability when needed. Valves isolating penetrations serving ESF can be operated from the control room to isolate an ESF System line when requried.

For further discussion, see the following section of the UFSAR:

Containment Isolation System	6.2.4

The fuel transfer tube is not a Containment penetration that qualifies as a fluid system penetration. The blind flange and in-Containment portion of the transfer tube are an extension of the Containment boundary. The blind flange isolates the transfer tube at all times except when the reactor is shut down for refueling. This assembly is a penetration in the same sense as are equipment hatches and personnel locks.

3.1.2.5.6 Criterion 55 – Reactor Coolant Pressure Boundary Penetrating Containment: Each line that is part of the RCPB and that penetrates the primary 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:

1. One locked-closed isolation valve inside and one locked-closed isolation valve outside the Containment.

2. One automatic isolation valve inside and one locked-closed isolation valve outside the Containment.

3. One locked-closed isolation valve inside and one automatic isolation valve outside the Containment. A simple check valve may not be used as the automatic isolation valve outside the Containment.

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

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

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

3.1.2.5.6.1 Evaluation Against Criterion 55 – Each line that is apart of the RCPB and penetrates the Containment is provided with isolation valves meeting this criterion. Instrument lines are designed in accordance with the requirements of RG 1.11.

3.1.2.5.7 Criterion 56 – Primary Containment Isolation: Each line that connects directly to the Containment atmosphere and penetrates primary Reactor Containment shall be provided with Containment isolation valves as follows, unless it can be demonstrated that the Containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked-closed isolation valve inside and one locked-closed isolation valve outside the Containment.

2. One automatic isolation valve inside and one locked-closed isolation valve outside the Containment.

3. One locked-closed isolation valve inside and one automatic isolation valve outside the Containment. A simple check valve may not be used as the automatic isolation valve outside the Containment.

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

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

3.1.2.5.7.1 Evaluation Against Criterion 56 – Each line that connects directly to the Containment atmosphere and penetrates the Containment is provided with Containment isolation valves, except where it can be demonstrated that the Containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable.

Lines which connect directly to the Containment atmosphere and penetrate the primary Containment are provided with two barriers in series, one inside and on outside, where they penetrate the Containment, so that failure of one barrier will not prevent isolation. The chilled water return lines from the RCFCs, are provided with two automatic isolation valves, both outside containment, to allow continued use of the penetration for CCW water to the RCFCs, an essential function. The double barrier provides greater reliability and eliminates potential leakage paths.

The isolation system for each line is designed to fail in a safe mode. Air-operated valves are designed to fail in the direction of greatest safety.

Motor-operated valves fail in the position which they are in when failure occurs. Different power sources for each valve in series ensure that isolation is not prevented by a single failure.

For further discussion, see the following sections of the UFSAR:

Containment Isolation Systems	6.2.4
Engineered Safety Features Actuation System	7.3
Accident Analyses	15.0

3.1.2.5.8 Criterion 57 – Closed System Isolation Valves: Each line that penetrates primary Reactor Containment and is neither part of the RCPB nor connected directly to the Containment atmosphere shall have at least one Containment isolation valve which shall be either automatic, locked-closed or capable of remote-manual operation. This valve shall be outside the Containment and located as close to the Containment as practical. A simple check valve may not be used as the automatic isolation valve.

3.1.2.5.8.1 Evaluation Against Criterion 57 – Each line that penetrates the Containment and is not connected directly to the Containment atmosphere and is not part of the RCPB has at least one isolation valve located outside Containment near the penetration. Details are provided in Section 6.2.4.

3.1.2.6 Group VI – Fuel and Radioactivity Control (Criteria 60-64).

3.1.2.6.1 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 operational limitations upon the release of such effluents to the environment.

3.1.2.6.1.1 Evaluation Against Criterion 60 – Waste handling systems are incorporated in the facility design for processing and/or retention of normal operation radioactive wastes. Controls and monitors capable of closing discharge isolation valves are provided to assure that releases are in accordance with NRC regulations as set forth in 10CFR20 and 10CFR50. The intent of these regulations is to ensure that the levels of any radioactive material effluents in unrestricted areas is as low as reasonably achievable.

The Liquid Waste Processing System (LWPS) is designed to recycle as much process waste as can be accommodated within the plant water balance. All releases are monitored and controlled and the system has been designed to prevent accidental discharges. The principal source of gaseous effluents from the plant during normal operation is the hydrogen continuously vented from the volume control tank. This gas is exhausted through an ambient temperature treatment system, including charcoal adsorbers, which removes radioiodines and particulates, to the plant main exhaust duct.

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

3.1.2.6.2 Criterion 61 – Fuel Storage and Handling and Radioactivity: The Fuel Storage and Handling System, Radioactive Waste System, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate Containment, confinement and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

3.1.2.6.2.1 New Fuel Storage – Evaluation Against Criterion 61 – New fuel is place in dry storage in the new fuel storage vault which is located inside the Fuel-Handling Building (FHB). The storage vault within the FHB provides adequate shielding for radiation protection. Storage racks preclude accidental criticality (see "Evaluation Against Criterion 62"). The new fuel storage racks do not require any special inspection and testing for nuclear safety purposes.

3.1.2.6.2.2 Spent Fuel Handling and Storage – Evaluation Against Criterion 61 – Irradiated fuel is stored underwater in spent fuel storage racks located at the bottom of the spent fuel pool. Spent fuel pool water is circulated through the Spent Fuel Pool Cooling and Cleanup System (SFPCCS) to maintain fuel pool water temperature, purity, water clarity, and water level. The spent fuel storage racks preclude accidental criticality (see "Evaluation Against Criterion 62").

Reliable decay heat removal is provided by the closed-loop SFPCCS, which consists of two cooling trains, two purification trains, a surface skimmer loop, and required piping, valves and instrumentation. Water is drawn from the spent fuel pool by the spent fuel pool pumps, is pumped through the tube side of the heat exchangers and is returned. Each suction line, which is protected by a strainer, is located at an elevation 4 ft below the normal water level, while the return line terminates in a sparger pipe at the bottom of the Spent Fuel Pool and contains an antisiphon hole near the surface of the water to prevent gravity drainage. The SFPCCS is designed to remove the amount of decay heat produced by the number of spent fuel assemblies that are stored following refueling. Each train is capable of removing 100 percent of the normal maximum design heat load and 50 percent of the abnormal maximum design heat load. Table 9.1-1 gives the peak SFP temperatures calculated for various fuel heat load and SFP cooling configurations.

System piping is arranged so that failure of any pipeline cannot drain the spent fuel pool or the in-Containment temporary storage area below a depth of approximately 23 ft of water over the top of the stored spent fuel assemblies. A minimum depth of approximately 13 ft of water over the top of an array of 193 (full core) assemblies with 42 hours of decay is required to limit radiation from the assemblies to 2.5 mR/hr. or less.

High- and low-level alarms in the control room are actuated upon pool water level changes. Fission product concentration in the pool water is minimized by use of the filters and demineralizers. This minimizes the fission product releases from the pool to the FHB environment.

Since SFP cooling is in continuous operation while irradiated fuel is stored in the pool (except during brief periods for testing and maintenance), and since the trains are frequently alternated to equalize run time, periodic surveillance tests are not required. Routine visual inspection of the system components, instrumentation and trouble alarms are adequate to verify system operability.

3.1.2.6.2.3 Radioactive Waste Systems – Evaluation Against Criterion 61 – The Radioactive Waste Systems provide all equipment necessary to collect, process and prepare for disposal all radioactive liquids, gases and solid waste produced as a result of operation.

The LWPS is divided into two sections: one section treats reactor-grade liquid which is recyclable after processing, and the other section treats non-reactor-grade water from inputs such as floor drains which is released from the plant after processing. Processing may include filtration, ion exchange, analysis, and evaporation. Spent resins are de-watered for disposal as solid radwaste. If conditions require, evaporator bottoms are processed for disposal as solid radwaste. Dry solid radwastes are packaged in steel drums or fiber drums, cartons or boxes. Gaseous radwastes are monitored, processed, recorded, and restricted so that radiation doses to members of the public in unrestricted areas are below those allowed by applicable regulations.

Routinely accessible portions of the FHB and Mechanical-Electrical Auxiliaries Building (MEAB) have sufficient shielding to maintain dose rates ALARA. See USAR, Section 12.3, for shielding design criteria. The MEAB and its associated systems are designed to preclude accidental release of radioactive materials to the environs.

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

The Fuel Storage and Handling and Radioactive Waste Systems are designed to assure adequate safety under normal and postulated accident conditions.

For further discussion, see the following UFSAR sections:

Residual Heat Removal System	5.4.7
Fuel Storage and Handling	9.1
Air Conditioning, Heating, Cooling, and Ventilation Systems	9.4
Radiation Protection	12.0
All Other Systems Required for Safety	7.6
Engineered Safety Features Actuation System	7.3
Radioactive Waste Management	11.0

3.1.2.6.3 Criterion 62 – Prevention of Criticality in Fuel Storage and Handling: Criticality in the Fuel Storage and Handling System shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

3.1.2.6.3.1 Evaluation Against Criterion 62 – Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality in the new fuel storage rack is prevented by the geometrically safe configuration of the storage rack. There is sufficient spacing between the assemblies to assure that the array when fully loaded is substantially subcritical. Criticality in the spent fuel storage rack is prevented by both the geometrically safe configuration of the storage rack and the use of administrative procedures to control the placement of burned and fresh fuel and control rod assemblies. Storage of fuel assemblies are limited by design to the loading and rack storage position.

New fuel and spent fuel storage rack design details are provided in Section 9.1.

The new fuel racks are designed to withstand nominal operating loads as well as SSE and Operating Basis Earthquake (OBE) seismic loads meeting Safety Class (SC) 3 and American Institute of Steel Construction (AISC) requirements. The new fuel racks are designed to withstand maximum uplift force of 5,000 pounds.

The center-to-center distance between the adjacent fuel assemblies is sufficient to ensure a $K_{eff} \leq 0.95$ even if unborated water is used to fill the new fuel storage area. The K_{eff} of the spent fuel storage racks is maintained as follows: a) less than or equal to 1.00 including uncertainties and tolerances on a 95/95 basis, even if unborated water is used to fill the spent fuel pool; and b) less than or equal to 0.95 including uncertainties, tolerances and accident conditions in the presense of spent fuel pool soluble boron.

Spent fuel racks are designed to withstand handling normal operating loads (impact and dead loads of fuel assemblies) as well as SSE and OBE seismic loads meeting SC 3 and AISC requirements. The spent fuel racks are also designed to meet Seismic Category I requirements of RG 1.13.

The spent fuel racks can withstand an uplift force equal to the uplift force of the spent fuel pool bridge hoist.

Refueling interlocks include circuitry which senses conditions of the refueling equipment. During refueling these interlocks reinforce operational procedures that prohibit making the reactor critical. The Fuel-Handling System is designed to provide a safe, effective means of transporting and handling fuel and is designed to minimize the possibility of mishandling or maloperation.

For further discussion, see the following UFSAR sections:

All Other Systems Required for Safety	7.6
Fuel Storage and Handling	9.1

3.1.2.6.4 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 RHR capability and excessive radiation levels and (2) to initiate appropriate safety actions.

3.1.2.6.4.1 Evaluation Against Criterion 63 – Appropriate systems are provided to meet the requirements of this criterion. A malfunction of the SFPCCS which would result in loss of RHR capability and excessive radiation levels is alarmed in the control room. Alarmed conditions include low flow on the refueling water purification pump, and high/low level in the spent fuel pool or in the in-Containment fuel storage area. The spent fuel pool water temperature and the in-Containment fuel storage area water temperature are continuously monitored and annunciated in the control room. The area Radiation Monitoring System continuously monitors radiation levels in these areas and alarms locally and in the control room at abnormal radiation levels.

Area radiation levels and tank levels are monitored and alarmed to give indication of conditions which may result in excessive radiation levels in radioactive waste system areas.

For further discussion, see the following UFSAR sections:

Process and Effluent Radiological Monitoring and Sampling System	11.5
Fuel Storage and Handling	9.1
Liquid Waste Management System	11.2
Gaseous Waste Management System	11.3
Solid Waste Management System	11.4

3.1.2.6.5 Criterion 64 – Monitoring Radioactivity Releases: Means shall be provided for monitoring the Containment atmosphere, spaces containing components for recirculation of LOCA fluids, effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

3.1.2.6.5.1 Evaluation Against Criterion 64 – The Containment atmosphere is continuously monitored during normal and transient operations, using the Containment particulate, iodine and gaseous monitors. Under postaccident conditions, samples of the Containment atmosphere will provide data on existing airborne radioactive concentrations within the Containment. Radioactivity levels contained in the facility effluent discharge paths and in the environs are continuously monitored during normal and accident conditions. The following potential station release paths are monitored:

- Unit Vent
- Secondary Side Steam Release
- Turbine Generator Building Drain Effluent
- Condenser Vacuum Pump Discharge

- Steam Generator Blowdown

Offsite monitoring is accomplished through the Radiological Environmental Monitoring Program (REMP).

The area Radiation Monitoring System monitors the plant environs for radiation. In addition to the fixed equipment, measurements are made using portable equipment.

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

Coolant Pressure Boundary Leakage Detection System	5.2.5
Process and Effluent Radiological Monitoring and Sampling Systems	11.5
Area Radiation Monitoring	12.3.4.1
Airborne Radioactivity Monitoring	12.3.4.2

REFERENCES

Section 3.1:

3.3-1 Mohan Thadani to James J. Sheppard, "South Texas Project, Units 1 and 2 - Issuance of Amendments - Re: Revision to Surveillance Requirement 3/4.6.2, 'Depressurization and Cooling Systems' (TAC Nos. MB9100 and MB9101);" August 20, 2003, (AE-NOC-03001108)

3.3-2 Robert Gramm to James J, Sheppard, "South Texas Project (STP), Units 1 and 2 - Re: Request for Relief from the Requirements of the American Society of Mechanical Engineers Boiler and Vessel Code (ASME code) (TAC Nos. MC0219 and MC0220);" March 26, 2004. (AE-NOC-04001224)

CN-3179

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

This material is discussed for balance-of-plant scope in Section 3.2.A and for Nuclear Steam Supply System scope in Section 3.2.B.

3.2.A Classification of Structures, Components, and Systems (Balance-of-Plant Scope)

Certain structures, components, and system of the nuclear plant are considered safety-related because they perform safety functions required to avoid or mitigate the consequences of abnormal operational transients or accidents. This section classifies structures, components, and systems according to the safety function they perform. In addition, design requirements are placed upon such equipment to assure the proper performance of safety actions, when required.

Chapter 13.7 describes alternate requirements for safety related non-risk significant and low safety significant structures, components, and systems.

3.2.A.1 Seismic Classifications. Safety-related plant structures, systems, and components are designed to withstand the effects of a Safe Shutdown Earthquake (SSE) (see Section 2.5) and remain functional[1] if they are necessary to assure:

1. The integrity of the reactor coolant pressure boundary (RCPB)

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

3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10CFR100

Plant structures, systems, and components, including their foundations and supports[2], that are designed to remain functional in the event of an SSE are designated as seismic Category I and are indicated in Table 3.2.A-1. These classifications meet the requirements of Regulatory Guide (RG) 1.29.

Structures, components, and systems designated as Safety Class (SC) 1, 2 or 3 (see Section 3.2.A.2 and American Nuclear Standards Institute [ANSI] N18.2a [1975] for a definition of safety classes) are generally classified as seismic Category I. For systems where postulated failure of components not designed for the SSE would result in conservatively calculated offsite exposures less than 0.5 rem, a nonseismic classification is assigned.

Components (and their supporting structures) which are not seismic Category I and whose collapse could result in loss of required function of structures, equipment or systems required after a SSE (e.g., through impact of flooding of seismic Category I structures) are analytically checked to confirm their integrity against collapse when subjected to seismic loading resulting from the SSE.

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[1] As defined by Regulatory Guide 1.29, positions 1 and 2.

[2] Piping supports are identified on Table 3.2.A-1 and 3.2.B-1 with the appropriate piping.

Seismic design of radwate management systems shall meet the intent of the simplified seismic and analysis procedure specified in Branch Technical Position (BTP) Effluent Treatment System Branch (ETSB) 11-1 (Rev. 1). Structures housing radwate management systems shall meet the intent of BTP ETSB 11-1 (Rev. 1).

All seismic Category I structures, systems, and components are analyzed under the loading conditions of the SSE or qualified by appropriate testing or by generic analysis. For further details of seismic design criteria, refer to the following sections:

Mechanical	3.7 and 3.9
Electrical	3.10
Structures	3.7 and 3.8
Instrumentation and Controls	3.10

An Operating Basis Earthquake, as defined in 10CFR100, Appendix A, is specified for those features of South Texas Project Electric Generating Station (STPEGS) necessary to remain functional for continued operation without undue risk to the health and safety of the public.

3.2.A.2 Quality Group Classification. Safety class terminology is utilized for the classification of components and structures for the STPEGS. This terminology correlates to the Nuclear Regulatory Commission (NRC) Quality Group designations for water, steam, and radioactive waste-containing mechanical components as follows:

STPEGS Classification	NRC RG 1.26
SC 1	Quality Group A
SC 2	Quality Group B
SC 3	Quality Group C
NNS (Non-Nuclear Safety)	Quality Group D

The safety classifications assigned to various components are tabulated in Table 3.2.A-1, and meet the requirements of RG 1.26 where applicable.

Structures, systems, and components are classified as SC 1, 2, 3, or NNS in accordance with the safety functions to be performed by such equipment. The importance of class assignment is considered in the design, for material selection, in manufacture or fabrication, and during assembly, erection, and construction.

Industry code requirements and Quality Assurance (QA) Program requirements for these systems, components, and structures are listed in Table 3.2.A-1.

The design requirements for equipment classified as NNS are specified with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate.

Where possible, design requirements are based on applicable industry codes and standards. Where these are not available, accepted industry or engineering practices are relied upon. Radioactive waste management systems which are classified as NNS are designed in accordance with the intent of BTP ETSB 11-1 (Rev. 1) where appropriate.

Equipment is assigned a specific SC recognizing that components within a system may be of differing safety importance. A single system may thus have components in more than one SC. Supports and restraints shall be in the same SC as the component supported if their failure could cause a loss of safety function of the associated supported component. Components with an NNS classification are located, protected, or supported so that their failure does not prevent safety-related components from performing their intended safety functions.

The design, fabrication, construction and testing of fire protection systems are performed in accordance with the applicable portions of the National Fire Protection Association Codes. QA program requirements ensure that the requirements for design, procurement, installation, testing, and administrative controls for the fire protection program are satisfied. The QA requirements applied to the fire protection program are in accordance with BTP Auxiliary Systems Branch (ASB) 9.5-1.

Fluid system safety classification and boundaries are indicated on the system piping and instrument diagrams in the respective section which describes the detailed design and safety analysis. The correlation of safety classification with industry codes and standards for mechanical components is found in Table 3.2.A-2. The following definitions apply to fluid system pressure boundary components and the Reactor Containment Building (RCB). A more complete definition of the SCs can be found in ANSI N18.2a-1975.

3.2.A.2.1 Safety Class 1: Safety Class 1 applies to components of the RCPB whose failure during normal reactor operations would prevent orderly reactor shutdown and cooldown assuming makeup water is provided by normal makeup systems only.

The code requirements, degree of QA, and seismic designations for SC 1 equipment are listed in Table 3.2.A-1.

3.2.A.2.2 Safety Class 2: SC 2 applies to the RCB and those components of the RCPB that are not SC 1, and to those components that are necessary to:

1. Remove residual heat directly from the reactor or reactor containment.

2. Circulate reactor coolant for any safety system purpose.

3. Control radioactivity release from within the RCB.

The code or standard requirements, the degree of QA, and the seismic designations for SC 2 equipment are listed in Table 3.2.A-1. Plant conditions and design loading combinations are detailed in Tables 3.9-2.2 and 3.9-2.3A.

3.2.A.2.3 Safety Class 3: SC 3 applies to those components that are not SC 1 or 2, but that are necessary to:

1. Provide or support any safety system functions.

2. Contain radioactive material other than radioactive waste management systems, and whose failure would release to the environment gaseous, liquid, or solid radioactivity resulting in a single-event whole-body dose greater than 0.5 rem at the site boundary.

3. Remove decay heat from spent fuel.

The code or standard requirements, the degree of QA, and seismic description for SC 3 mechanical equipment are listed in Table 3.2.A-1. Plant conditions and design loading combinations are detailed in Tables 3.9-2.2 and 3.9-2.3A.

3.2.A.2.4 Non-Nuclear Safety: NNS applies to portions of the nuclear power plant not covered by SC 1, 2, or 3 that can influence safe normal operation or that may contain radioactive fluids. These apply primarily to components of secondary systems and waste disposal systems not otherwise covered. Also included are safety system components whose failure would not degrade system performance or cause a release to the environment of gaseous activity normally required to be held for decay.

3.2.A.2.5 Not Applicable: Those systems not designated SC 1, SC 2, SC 3, or NNS which neither connect to nor influence equipment within the SCs defined above and do not contain radioactive fluids are designated not applicable (NA). SCs have not been defined for those systems other than fluid system pressure boundary components and the Containment; therefore, the SC designation for those systems is NA.

3.2.B Classification of Structures, Components, and Systems (Nuclear Steam Supply System [NSSS] Scope)

3.2.B.1 Seismic Classification. In lieu of the requirements of RG 1.26 and 1.29, Westinghouse Electric Corporation applies a rule that each component classified as SC 1, 2, or 3 shall be qualified to remain functional in the event of the SSE* except where exempted by meeting all of the following conditions. Portions of systems required to perform the same safety function required of any SC component which is a part of that system shall be likewise qualified or granted exemption. Provisions to be met for exemption are:

1. Failure would not directly cause a Condition III or IV event (as defined in Ref. 3.2.B-4).

2. There is no safety function to perform nor could failure prevent mitigation of the consequences of a Condition III or IV event.

* “Safe Shutdown Earthquake” is the same earthquake as the “Design Basis Earthquake” defined in N18.2-1973, Criterion 2.1.5.4.

3. Failure during or following any Condition II event would result in consequences no more severe than allowed for a Condition III event.

A further explanation of the above exemption criteria follows:

a. All SC 1 components must be seismically qualified since a failure on any one can directly cause a Condition III or IV event, thus failing Provision 1 for exemption.

b. SC 2 components that are part of the RCPB must be seismically qualified because of the rule stated above, "Portions of a system required to perform the same safety function required of a safety class component which is a part of that system shall be likewise qualified or granted exemption."

c. All other SC 2 components must also be seismically qualified because they are required to mitigate, or their failure could prevent mitigation of, the consequences of a Condition III or IV event, thus failing the second provision for exemption.

d. Components placed in SC 3 by reason of item (1), (3) or (4) of Criterion 2.2.3, the SC 3 definition of ANSI N18.2a, meet Provisions 1 and 2 for granting the seismic design exemption.

   Since exemption Provisions 1 and 2 are fulfilled, seismic qualification is not required if Provision 3 is met. The basis for judgment of this third provision is the rule of Criterion 2.1.3.3 on ANSI N18.2: "The release of radioactive material due to Condition III incidents may exceed guidelines of 10CFR20, ′Standards for Protection Against Radiation, ′ but shall not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius."

e. Components placed in SC 3 by reason of item (2) of Criterion 2.2.3 of ANSI N18.2 must be seismically qualified for the same reason as given in item c, above.

Chapter 13.7 describes alternate requirements for safety related non-risk significant and low safety significant structures, components, and systems.

Table 3.2.B-1 shows the SC, the code class, the degree of QA, and the method of seismic qualification for the listed components. Seismically qualified components are qualified to remain functional in the event of the SSE, as defined above.

Table 3.2.B-2 shows the applied Westinghouse Electric Corporation equivalent SCs for non-fluid system components, with explanations for the choices.

3.2.B.2 System Quality Group Classifications. Components are classified as SC 1, 2, or 3 or NNS in accordance with ANSI N18.2a-1975 classification (Ref. 3.2.B-3). This classification system is compatible with the requirements of RG 1.26.

Classification of piping and valves, and interfaces from one SC to another, are shown on the system piping and instrument diagrams shown elsewhere in this safety analysis report.

3.2.B.3 Equipment Code and Classification List. Table 3.2.B-1, “Equipment Code and Classification List-Westinghouse Fluid System Components,” tabulates components by SC assignment and applicable code class in accordance with Ref. 3.2.B-5. The earliest applicable code for the pressure vessels which are part of the RCPB is the 1971 edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, with application of all addenda through, to, and including the Winter 1972 addenda. The earliest applicable code for pumps and valves which are part of the RCPB is the 1971 edition, with application of all addenda through, to, and including the Winter 1972 addenda. The earliest applicable code for the piping which is part of the RCPB is the 1971 edition, with application of all addenda through, to, and including the Summer 1973 addenda. Later code editions may be used optionally.

As indicated in the footnotes to Table 3.2.B-1, some NSSS components are built to a more stringent design code than required.

3.2.C Use of USNRC Generic Letter 89-09 for ASME Component Replacements.

Generic Letter 89-09 provides guidance for purchasing replacements which are no longer available in full compliance with the stamping and documentation requirements of Section III of the Code. STPEGS has utilized similar relief requested by ST-HL-AE-1156 dated February 25, 1985 and accepted by USNRC letter dated May 21, 1985.

3.2.C.1 Generic Letter 89-09 Replacement List. Table 3.2.C-1 “Generic Letter 89-09 Replacement List,” tabulates the replacement items purchased utilizing the guidance of Generic Letter 89-09 by item description, purchase order, vendor, part number and intended/potential application(s).

REFERENCES

Section 3.2:

3.2.B-1 Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants".

3.2.B-2 Regulatory Guide 1.29, "Seismic Design Classification".

3.2.B-3 ANSI N18.2a-1975, "Revision and Addendum to Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants".

3.2.B-4 ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants".

3.2.B-5 ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Components".

TABLE 3.2.A-1

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Reactor Coolant System					See P&IDs, Section 5.2
Piping, supports and valves	1, 2, NNS	III/1, III/2 ANSI B31.1	I, NA	B, NA	See Note 8
Chemical and Volume Control System					See P&IDs, Section 9.3.4
Boric acid tanks	3	III/3	I	B	
Pulsation dampeners	2	III/2	I	B	
Piping, and valves and supports, etc.	2, 3, NNS	III/2, III/3 ANSI B31.1	I, NA	B, NA	See Note 8
Residual Heat Removal System					See P&IDs, Section 5.4
Piping, valves and supports, etc.	1, 2	III/1, III/2	I	B	See Note 8
Spent Fuel Pool Cooling and Cleanup System					See P&Ids, Section 9.1.3
Piping, supports and valves in cooling loops	3	III/3	I	B	
Piping, supports and valves in purification and in skimming loop	NNS	ANSI B31.1	NA	NA	
Piping and supports for Containment Isolation	2	III/2	I	B	

* See Notes at end of table for abbreviations

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class$^{(1)}$	Standard Or Code$^{(2)}$	Seismic Category$^{(3)}$	Quality Assurance$^{(4)}$	Remarks
Containment Spray System					See P&IDs, Section 6.2.2
Piping valves and supports, etc.	2, 3, NNS	III/2, III/3, ANSI B31.1	I, NA	B, NA	See Note 8
Boron Recycle System					See P&IDs, Section 9.3
Recycle holdup tanks	NNS	API-650, AWWA-D-100, or ANSI B96.. 1	NA	a	See Note 11
Pipes, valves, and supports	NNS	ANSI B31.1	NA, a	a	See Note 11
Reactor Makeup Water System (RMWS)					See P&IDs, Section 9.2.7
RMW storage tank	3	III/3	I	B	
RMW pumps	3	III/3	I	B	
Piping, supports, and valves, etc.	3, NNS	III/3, ANSI B31.1	I, NA	B, NA	See Note 8
Makeup Demineralizer System					See P&IDs See 9.2.3
Containment penetration, supports, and isolation valves	2	III/2	I	B	
Remainder of system	NNS	ANSI B31.1	NA	NA	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class[1]	Standard Or Code[2]	Seismic Category[3]	Quality Assurance[4]	Remarks
Main Steam (MS) System					See P&IDs Section 10.3 See Note 13
Those portions of the MS System including supports extending from and including the secondary side of SGs up to and including the first isolation valve and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation	2	III/2	I	B	
Code safety valves	2	III/2	I	B	
MS drain lines and valves to outside of isolation valve cubicle wall	2	III/2	I	B	
Power-operated relief valves	2	III/2	I	B	
Remainder of system	NNS	ANSI B31.1	NA	NA	See Note 8
Steam supply to turbine driven auxiliary feedwater pump, along with associated vents and drains	2, 3	III/2, 3	I	B	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Steam Generator Blowdown System					
The portion of the steam generator blowdown system including supports extending from the secondary side of the steam generators up to and including the first isolation valve on the main blowdown lines and connecting piping up to and including the second valve on the sample lines.	2	III/2	I	B	See P&IDs, Section 10.4.8 See Note 13
Flash tank	NNS	VIII	NA	a	
Recirculation pump	NNS	HI	a	a	
SGB regenerative heat exchanger	NNS	VIII, TEMA	a	a	
Mixed bed demineralizers	NNS	VIII	a	a	
Remaining piping, supports, and valves	NNS	ANSI B31.1	NA	NA	See Note 8
Component Cooling Water System					For more detail, refer to P&IDs, Section 9.2.2
Heat exchangers	3	III/3	I	B	
Pumps	3	III/3	I	B	
Surge tank	3	III/3	I	B	
Chemical addition tank	NNS	VIII	NA	NA	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Component Cooling Water System (Cont'd)					
Piping, supports, and valves for containment isolation	2	III/2	I	B	
Piping, supports, and valves other than those required for containment isolation or servicing NNS equipment	3	III/3	I	B	
Vent and drain piping and supports up to and including first isolation valve except in NNS portions	3	III/3	I	B	See Note 5
Piping, supports, and valves serving NNS equipment including vent and drain piping and supports	NNS	ANSI B31.1	NA	NA	See Note 5
Essential Cooling Water System (ECWS)					For more detail, refer to P&IDs , Section 9.2.1.2
Essential cooling water pumps	3	III/3	I	B	
ECW self-cleaning strainer	3	III/3	I	B	
ECW pump lubrication strainers	3	III/3	I	B	
Screen wash pumps	3	III/3	I	B	
Traveling screens	3	MS	I	B	
Piping and supports	3, NNS	III/3, ANSI B31.1	I, NA	B, NA	See Notes 5 and 8

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class[(1)]	Standard Or Code[(2)]	Seismic Category[(3)]	Quality Assurance[(4)]	Remarks
Essential Cooling Water System (ECWS) (Cont'd)					
Valves	3	III/3	I	B	
Auxiliary Feedwater (AFW) System					Fore more detail, refer to P&IDs, Section 10.4.9
Pumps	3	III/3	I	B	
AFW pump turbine	3	Built III/3	I	B	Not N-stamped
AFW piping and supports from AFWST to first isolation valve outside Containment	3	III/3	I	B	
AFW steam line and supports from main steam line isolation valves (steam inlet and steam inlet bypass valves) to AFW pump turbine and exhaust line	3	III/3	I	B	
Containment isolation valves, penetrations, AFW piping and supports from the isolation valves to SG	2	III/2	I	B	
AFW pump test/recirculation lines inside IVC also AFW cross connecting piping and valves	3	III/3	I	B	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Auxiliary Feedwater (AFW) System (Cont'd)					
Main steam isolation valves, (steam inlet and steam inlet bypass valves), AFW steamline and supports upstream of the isolation valve	2	III/2	I	B	
AFW storage tank (FWST) except liner	3	ACI 318-71, AISC-69	I	B	Concrete tank
AFWST liner	3	ASME III, AISC-69	I	B	Not N-stamped
Remainder of system	NNS	ANSI B31.1	NA	NA	
Feedwater (FW) System					For more detail, refer to P&IDs, Section 10.4.7
Those portions of the FW System including supports extending from and including the secondary side of SGs up to and including the first isolation valve and connected piping up to and including the first valve that is capable of automatic closure during all modes of normal reactor operation or the second normally closed valve	2	III/2	I	B	See Note 13
Remainder of system	NNS	ANSI B31.1	NA	NA	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class[(1)]	Standard Or Code[(2)]	Seismic Category[(3)]	Quality Assurance[(4)]	Remarks
Boron Thermal Regeneration System					See P&IDs Section 9.3.4
Piping, valves and supports, etc.	NNS	ANSI B31.1	NA	NA	See Note 11
Emergency Core Cooling System (Safety Injection System)					See P&IDs Section 6.3
Piping, valves and supports	1, 2, NNS	III/1, III/2, ANSI B31.1	I, NA	B, NA	See Note 8
Refueling water storage tank	2	III/2	I	B	
Sampling System					For more detail, see P&IDs, Section 9.3.2
Sample HXs	NNS	VIII	NA	NA	
Sample vessels	NNS	VIII	NA	NA	
Sample delay coil	2	III/2	I	B	Constructed of large diameter piping
Piping, supports, penetrations and valves inside Containment, up to and including first isolation valve outside Containment (samples originating inside Containment)	2	III/2	I	B	See Note 5 and 8

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class[(1)]	Standard Or Code[(2)]	Seismic Category[(3)]	Quality Assurance[(4)]	Remarks
Sampling System (Cont'd)					
Piping, supports, and valves downstream of the sample line isolation or root valve	NNS	ANSI B31.1	NA	NA	See Note 8 (Class break at root valve, which is part of sampled system.)
Reactor Coolant Vacuum Degassing System					See P&ID, Section 11.3 See Note 6
Gas storage tanks	NNS	VIII	NA, a	a	
Vacuum pump package	NNS	MS	NA, a	a	
Compressor package	NNS	MS	NA, a	a	
Remainder of piping, supports, and valves	NNS	ANSI B31.1	NA, a	a	
Containment penetration, piping, supports and valves	2	III/2	I	B	
Gaseous Waste Processing System					For more detail see P&IDs, Section 11.3
Charcoal beds and guard bed	NNS	VIII	NA, a	a	See Notes 6 and 11
Filter (HEPA)	NNS	MS	NA, a	a	
BRS recycle holdup tank vent compressor	NNS	MS	NA, a	a	
Control panels	NNS	MS	NA	NA	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Gaseous Waste Processing System (Cont'd)					
Remainder of piping supports and valves	NNS	ANSI B31.1	NA, a	a	
Liquid Waste Processing System					For more detail, see P&IDs, Section 11.2
Heat Exchanges	NNS	VIII	NA, a	a	
Tanks	NNS	VIII, API 620, 650	NA, a	a	
Resin fill tank	NNS	VIII	NA	NA	
Evaporators	NNS	VIII	NA, a	a	
Pumps	NNS	MS	NA, a	a	
Demineralizers	NNS	VIII	NA, a	a	
Control panels	NNS	MS	NA	NA	
Containment isolation valves, piping and supports	2	III/2	I	B	
Remainder of piping, supports and valves	NNS	ANSI B31.1	NA, a	a	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Radioactive Vents and Drains					See P&IDs, Section 9.3.3
Containment isolation valves, piping and supports	2	III/2	I	B	
Remainder of system	NNS	ANSI B31.1	NA, a	a	See Note 5
Solid Waste Processing System					See P&IDs, Section 11.4
Drumming Station Equipment:					
Baler	NNS	MS	NA, a	a	
Overhead crane	NA	CMAA-70	Note 6	d	
Drum shields	NNS	MS	NA, a	a	
Pumps	NNS	MS	NA, a	a	
Mixing tank	NNS	API 650	NA, a	a	
Cement tanks	NNS	MS	NA, a	a	
Waste-cement mixer	NNS	MS	NA, a	a	
Cement feeder	NNS	MS	NA, a	a	
Piping, supports and valves	NNS	ANSI B31.1	NA, a	a	
Control panel	NNS	MS	NA	NA	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Diesel Generator Lube Oil System					See P&IDs, Section 9.5.7
Piping, supports and valves up to and including normally closed valves	3	III/3	I	B	
Remainder of system	NNS	ANSI B31.1	NA	NA	
Diesel Generator Fuel Oil Storage and Transfer System					See P&IDs, Section 9.5.4
Diesel oil storage tanks	3	III/3	I	B	
Valves between storage tanks and engine	3	III/3	I	B	
Piping, valves and supports except vent and fill piping upstream of the locked closed valve	3	III/3	I	B	
Vent and fill piping, and supports upstream of the locked closed valve	NNS	ANSI B31.1	NA	NA	
Remainder of system	3, NNS	III/3 ANSI B31.1	I, NA	B, NA	See Note 8

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Diesel Generator Cooling Water System					See P&IDs, Section 9.5.5 See Note 5
Piping, supports and valves upstream of solenoid valve	NNS	ANSI B31.1	NA	NA	
Piping, supports and valves downstream of solenoid valve including the valve	3	III/3	I	B	
Diesel Generator Air Starting System					For more detail see Section 9.5.6
Air Dryers	NNS	MS	NA	NA	
Compressors	NNS	MS	NA	NA	
Air receivers	3	III/3	I	B	
Piping, valves, and supports from air compressor to the air receiver inlet check valve	NNS	ANSI B31.1	NA	NA	
Piping, valves and supports from the air receiver inlet check valve to the diesel generator skid	3	III/3	I	B	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Diesel Generator Intake and Exhaust System					See P&IDs, Section 9.5.8
Exhaust silencer	NNS	MS	I	e	
Inlet silencer	NNS	MS	I	e	
Inlet air filter	NNS	MS	I	e	
Expansion joints					
Inlet	NNS	MS	I	e	
Exhaust	NNS	MS	I	e	
Combustion air intake piping and supports	**	**	I	e	Not N-stamped
Diesel exhaust piping and supports	**	**	I	e	Not N-stamped
Containment Hydrogen Monitoring System					Fore more detail, see P&IDs, Section 7.6.5
Hydrogen analyzer package	NA	IEEE 279	I	B	
Piping, supports, and valves	2	III/2	I	B	Note 5
Sample vessels	NNS	VIII	NA	NA	

** Purchased to ASME III requirements, seismically designed, installed to B31.1, and welded to ASME Section IX.

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class[1]	Standard Or Code[2]	Seismic Category[3]	Quality Assurance[4]	Remarks
Fire Protection System					See P&IDs, Section 9.5.1 and Appendix 9.5.1.A
Containment penetration supports and isolation valves	2	III/2 MC, CC-3000	I	B	
Water supply system	NNS	NFPA 20 22, & 24	NA	b	
Fixed water spray deluge	NNS	NFPA 15	NA	b	
Sprinkler (automatic and pre-action)	NNS	NFPA 13	NA	b	
Standpipe (wet and pre-action)	NNS	NFPA 14	NA	b	
Foam extinguishers	NNS	NFPA11	NA	b	
Foam water sprinkler	NNS	NFPA 16	NA	b	
Halon 1301	NNS	NFPA 12A	NA	b	
Fire protection detection, control and annunciation	NNS	NFPA 72D & 72E	NA	b	
Station Air System					See P&IDs, Section 9.3.1
Containment penetrations supports and isolation valves	2	III/2	I	B	
Remainder of system	NNS	ANSI B31.1	NA	NA	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class[(1)]	Standard Or Code[(2)]	Seismic Category[(3)]	Quality Assurance[(4)]	Remarks
Instrument Air System					See P&IDs, Section 9.3.1
Containment penetrations, supports an isolation valves	2	III/2	I	B	
Remainder of system	NNS	ANSI B31.1	NA	NA	
Breathing Air System					
Containment penetrations, supports and isolation valves	2	III/2	I	B	
Remainder of system	NNS	ANSI B31.1	NA	NA	
HVAC					
1. Containment Building HVAC					For system classification boundaries, refer to Figures 9.4.5-1 through 9.4.5-3
RCFCs cooling coils	2	III/2	I	B	
RCFC ductwork	3	X	I	B	See Note 17
RCFC dampers	2	X	I	B	See Note 17
RCFC fans	2	X	I	B	
Containment cubicle exhaust fans, ductwork and dampers	3	X	I	B	
Containment carbon units	NNS	X, Z	NA	NA	See Note 6

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class[1]	Standard Or Code[2]	Seismic Category[3]	Quality Assurance[4]	Remarks
HVAC (Cont'd)					
Normal purge supply and exhaust fans, filters, ductwork, and dampers	NNS	X, Z	NA	NA	See Note 6
Containment purge isolation valves, penetrations and supports	2	III/2	I	B	
Supplementary purge supply and exhaust fans, filters, ductwork, and dampers	NNS	X, Z	NA	NA	See Note 6
Reactor cavity and support supply and exhaust fans, ductwork and dampers	NNS	X	NA	NA	See Note 6
Tendon access gallery ventilation fans and ductwork	NNS	X	NA	NA	
Tendon access gallery tornado dampers	3	X	I	B	
Chillers	NNS	VIII, Y	NA	NA	
Chilled water pumps	NNS	VIII	NA	NA	
Chilled water piping and supports, etc.	NNS	III/2 ANSI B31.1	NA	NA	
Air separators	NNS	ANSI B31.1	NA	NA	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class[1]	Standard Or Code[2]	Seismic Category[3]	Quality Assurance[4]	Remarks
HVAC (Cont'd)					
2. Mechanical Auxiliary Building HVAC					See P&IDs, Section 9.4.3
Fans, ductwork and dampers for supplementary coolers	NNS	X	NA	NA	See Note 6 for except ductwork
Filters	NNS	Z	NA	NA	
Expansion tank and chemical addition tank	NNS	VIII	NA	NA	
Tornado dampers	3	X	I	B	
Isolation dampers from the MAB outside air intake to the EAB	NNS	X	NA	NA	
Cooling Coils except for supplementary coolers	NNS	VIII	NA	NA	
Chiller	NNS	VIII, Y	NA	NA	
Chilled water pumps	NNS	VIII	NA	NA	
Chiller water piping, supports and valves	NNS	ANSI B31.1	NA	NA	
Supplementary coolers including ductwork cooling coils and fans	3	III/3, X	I	B	
Supplementary fan cooler units	3	III/3, X	I	B	
Air separators	NNS	ANSI B31.1	NA	NA	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
HVAC (Cont'd)					
3. Control Room and Electrical Auxiliary Building HVAC					See P&IDs, Section 9.4.1
Air handling units	3	X, Y, Z, III/3	I	B	
Fans, ductwork, and dampers except exhaust fans	3	X	I	B	See Note 17
Electrical penetration area, supplementary coolers, fans, coils, and ductwork (normal) (emergency)	NNS 3	X III/3, X	NA I	NA B	See Note 6 See Note 17
EAB battery room heating coil	3	X	I	B	Safety-related battery rooms only.
Other heating coils	NNS	X	NA	NA	See Note 6
Battery room exhaust fans	3	X	I	B	Safety-related battery rooms only.
Other exhaust fans (toilet/kitchen)	NNS	X	NA	NA	See Note 6
Filters	3	Z	I	B	
Chillers	3	Y, III/3	I	B	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
HVAC (Cont'd)					
Chilled water pumps	3	III/3	I	B	
Chilled water piping, supports and valves	3	III/3	I	B	
Chilled water expansion tank	3	III/3	I	B	See Note 5
Chilled water chemical addition tank	NNS	VIII	NA	NA	
Penetration space exhaust fans, ductwork, and dampers	NNS	X	NA	NA	See Note 6
4. Fuel-Handling Building HVAC					For systems classification boundaries, refer to Figures 9.4.2-1 and 9.4.2-2
Supply fans, ductwork, and dampers (except safety portions)	NNS	X	NA	NA	See Note 6
Main exhaust fans, exhaust booster fans, ductwork, and dampers	3	X	I	B	See Note 17
Supply filters	NNS	Z	NA	NA	
Exhaust filters	3	Z	I	B	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
HVAC (Cont'd)					
Supply cooling coils	NNS	VIII, Y	NA	NA	
Supplementary coolers including ductwork and cooling coils (except post-accident sampling station area cooler)	3	III/3, X	I	B	See Note 17
Piping, supports and valves	3, NNS	III/3, ANSI B31.1	I, NA	B, NA	See Note 8
Intake ductwork including emergency make-up dampers up to the supply damper to each AHU	3	X	I	B	See Note 17
Post-accident sampling station area supplementary cooler including ductwork and cooling coils	NNS	Y	NA	NA	
5. Diesel Generator Building HVAC					See P&IDs, Section 9.4.6
Emergency fans, ductwork, and dampers	3	X	I	B	See Note 17
Normal fans, ductwork, and filters	NNS	X	NA	NA	See Note 6

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
HVAC (Cont'd)					
6. Essential Cooling Water Intake Structure HVAC					See P&IDs, Section 9.4.7
Fans, ductwork, and dampers	3	X	I	B	See Note 17
7. Turbine Generator Building HVAC System					See P&IDs, Section 9.4.4
Fans, filters, ductwork, and dampers	NNS	X, Z	NA	NA	
Package AC units	NNS	X, Y	NA	NA	
8. MSIV Building HVAC					For system classification boundaries refer to Figure 9.4.8-1. See Note 17
MSIV fans and ductwork	3	X	I	B	
Restraints/Penetration	NNS	X	NA	NA	See Note 6
9. Technical Support Center HVAC					See P&IDs, Section 9.4.1
Supply and exhaust fans, ductwork and dampers	NNS	X	NA	NA	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
HVAC (Cont'd)					
Supply filters	NNS	Z	NA	NA	
Chillers	NNS	VIII, Y	NA	NA	
Chilled water pump	NNS	VIII	NA	NA	
Chilled water expansion, chemical addition tanks	NNS	VIII	NA	NA	
Piping, supports and valves	NNS	ANSI B31.1	NA	NA	
Cranes					
Polar crane	NA	CMAA 70 (Note 18)	Note 6	d	
Cask-handling overhead crane	NA	CMAA 70	Note 6	d	
Fuel-Handling Building overhead crane	NA	CMAA 70	Note 6	d	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Structures					
Reactor Containment Building	2	ASME/ACI 359 AISC-69 ACI 318-71	I	B	See Section 3.8
Equipment hatch	2	III/MC	I	B	N-stamp not required
Airlocks	2	III/MC	I	B	
Penetration sleeves	2	III/MC, CC-3000	I	B	N-stamp not required
Containment mechanical penetration sleeves	2	III/MC	I	B	N-stamp not required
Liner plate	2	III/2 CC-3000	I	B	N-stamp not required
Containment coatings	3	ANSI N101.2	NA	C	
Containment internal structures	NA	ACI 318-71 AISC-69	I	B	
Pipe whip restraints, barriers for pipe break and missile hazard protection (if not included in above)	NA	AISC-69 ACI 318-71	I	B	
Reactor Containment fan cooler structure	NA	ACI 318-71 AISC-69	I	B	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Reactor Containment Building (Cont'd)					
Supports for seismic Category I equipment	Note 9	AISC-69 ACI 318-71	I	B	
Mechanical-Electrical Auxiliaries Building, including Control Room	NA	ACI 531-79 AISC-69 ACI 318-71	I	B	See Section 3.8
Supports for seismic Category I equipment	Note 9	AISC-69 ACI 318-71	I	B	
Internal missile barriers and whip restraints	NA	AISC-69 ACI 318-71	I	B	
Fuel-Handling Building	NA	ACI 531.79 AISC-69 ACI 318-71	I	B	See Section 3.8
Supports for seismic Category I equipment	Note 9	AISC-69 ACI 318-71	I	B	
Supports for crane	NA	AISC-69 ACI 318-71	Note 6	NA	
Internal missile barriers	NA	AISC-69 ACI 318-71	I	B	
Spent fuel pool and liner	NA	AISC-69 ACI 318-71	I	B	
High Density Spent Fuel Racks	3	ASME III	I	B	See Section 9.1.2

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Diesel Generator Building	NA	ACI 531-79 AISC-69 ACI 318-71	I	B	See Section 3.8
Supports for seismic Category I equipment	Note 9	AISC-69 ACI 318-71	I	B	
Internal missile barriers	NA	AISC-69, ACI 318-71	I	B	
Miscellaneous Structures					See Secs. 3.8 and 3.10
Essential cooling water intake and discharge structures including internal missile barriers (Safety-related portions)	NA	ACI 318-71	I	B	
Essential cooling pond	NA	NA	I	B	(North embankment excepted)
MSIV Structure including internal missile barriers	NA	ACI 318-71 AISC 69	I	B	
FW valve structures	NA	ACI 318-71	I	B	
Class 1E underground ductbank raceway system	NA	ACI 318-71 IEEE 344	I	B	See Secs. 3.8 and electrical 3.10

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Special doors (all structures)	NA	NA	I	B	(water tight, and missile resistant)
Structural backfill adjacent to Category I structures and under ECW piping	NA	NA	I	B	
Class 1E Electrical System Components					
For systems classification boundaries, refer to the one-line diagrams, Figures 8.3-1 and 8.3-3.					
4,160 V switchgear (ESF buses)	NA	IEEE 323, 344, 383, 384	I	B	
4,160/480 V transformers (ESF load centers)	NA	IEEE 323, 344, 383	I	B	
480/120/208 V transformers (control room and ESF area lighting)	NA	IEEE 323, 344	I	B	
480 V switchgear (ESF load centers)	NA	IEEE 323, 344, 383, 384	I	B	
480 V motor control and MCCs (ESF MCCs)	NA	IEEE 323, 344, 383, 384	I	B	
125 V station batteries and racks (control and vital instrumentation power supplies)	NA	IEEE 323, 344, 450, 485, 535	I	B	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Class 1E Electrical System Components (Cont'd)					
480 vac/125 vdc battery chargers (for vital dc bus)	NA	IEEE 323, 344, 383, 384, 650	I	B	
125 vdc switchboards and panels (vital dc power distribution)	NA	IEEE 323, 344, 420, 383, 384	I	B	
Class 1E inverters	NA	IEEE 323, 344, 650	I	B	
Voltage regulators (backup for instrumentation inverters)	NA	IEEE 323, 344, 383, 384	I	B	
120 vac instrument bus panels (vital instrumentation ac power distribution)	NA	IEEE 323, 344, 420, 383, 384	I	B	
Containment penetration assemblies (electrical portions)	2	IEEE 317, 323, 344, 383	I	B	
Diesel generator and accessories	NA	MS, IEEE 323, 344, 387	I	B	
Diesel generator control panels	NA	IEEE 323, 344, 383, 420	I	B	
Relay boards and racks	NA	IEEE 323, 344, 383, 384, 420	I	B	
Termination cabinets	NA	IEEE 323, 344, 383	I	B	
Wire and cable raceway system except conduit	NA	IEEE 344	I	B	See Note 17

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class[1]	Standard Or Code[2]	Seismic Category[3]	Quality Assurance[4]	Remarks
Class 1E Electrical Component Systems (Cont'd)					
Underground electrical duct bank system	NA	IEEE 344	I	B	
Cable system (power, control and instrumentation)	NA	IEEE 323, 383	I	B	
Electrical supports (for 1E systems)	NA	IEEE 344	I	B	See Note 17
Motors (for class 1E components)	NA	IEEE 323, 334, 344	I	B	
Valve operators for safety related valves	NA	IEEE 323, 344, 382, 383	I	B	
Conduit	NA	NA	NA	NA	See Note 6
Instrumentation & Control System Components					
Engineered safety feature (ESF) actuation systems (Non-NSSS portion)	NA	IEEE 279, 323, 344	I	B	
ESF isolation devices	NA	IEEE 279, 323, 344	I	B	
Radiation monitoring system (safety-related and RG 1.97 components)	NA	IEEE 279, 323, 344	I, NA	B	See Notes 12 and 14
Radiation monitoring system (RCPB monitors)	NA	IEEE 344	Note 10	NA	

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class(1)	Standard Or Code(2)	Seismic Category(3)	Quality Assurance(4)	Remarks
Instrumentation & Control System Components (Cont'd)					
Seismic instrumentation	NA	NA	Note 10	NA	
Systems required for safe shutdown	NA	IEEE 279, 323, 344	I	B	
Post-Accident monitoring system	NA	IEEE 279, 323, 344	I, NA	See Note 12	
Auxiliary shutdown panels and transfer switch panels	NA	IEEE 279, 323, 344	I	B	
Safety-related instrument tubing and fittings	2, 3	III/2, III/3	I	B	See Note 8
Qualified Display Processing System	NA	IEEE 279, 323, 344	I	B	See Note 12
Safety-related process instruments	NA	IEEE 279, 323, 344	I	B	See Note 8
Meteorological monitoring system	NA	RG 1.23, NUREG-0654	NA	f	
Loose parts monitoring system	NA	MS	***	NA	
ESF status monitoring system	NA	MS	NA	NA	

*** To remain functional after an OBE.

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structure, System Components	Safety Class[1]	Standard Or Code[2]	Seismic Category[3]	Quality Assurance[4]	Remarks
Control panels	NA	IEEE 323, 344, 383, MS, NA	I, NA	B, NA	
Main control board	NA	IEEE 323, 344, 383, 420	I	B	Only that portion of the main control board relating to Class 1E instruments and equipment.
ESF load sequencers	NA	IEEE 323, 344, 279, 383	I	B	
RCPB leak detection instruments	NA	NA	Note 10	NA	
Post-Accident Sampling System					
Piping, supports and valves up to and including the outside Containment isolation valve	2	III/2	I	B	
Remainder of system	NA	MS, ANSI B31.1	NA	NA	
Auxiliary Steam System					
MAB temperature element, isolation valves, piping, and supports	3	III/3	I	B	See P&ID's, Section 9.5.9
Remainder of system	NNS	ANSI B31.1	NA	NA	

TABLE 3.2.A-1

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

NOTES

1.	1, 2, 3, NNS	=	Safety classes defined in ANSI N18.2a.
	NA	=	Not applicable
2.	III/I	=	ASME B&PV Code, Section III, Code Class 1
	III/2	=	ASME B&PV Code, Section III, Code Class 2
	III/3	=	ASME P&PV Code, Section III. Code Class 3
	VIII	=	ASME B&PV Code, Section VIII
	MS	=	Manufacturer's standards
	X	=	The following shall apply: a. ASHRAE Guide (American Society of Heating, Refrigerating and Air Conditioning Engineers) b. AMCA 300-67, 210-74 (Air Moving and Conditioning Association) c. SMACNA Duct Construction Manual (Sheet Metal and Air Conditioning Contractors National Association)
	Y	=	ARI 410 Standard for Forced Circulation of Air Cooling and Heating Coils, 550, Standard for Centrifugal Water Chilling Packages, 590, Standard for Reciprocating Water Chilling Packages, (Air Conditioning and Refrigeration Institute)
	Z	=	The following shall apply: a. ASHRAE 52, Methods of Testing Air Cleaning Devices used in General Ventilation for Removing Particulate Matter b. UL 900, Air Filter Units (Underwriter's Laboratory)
	NFPA	=	National Fire Protection Association
	CMAA	=	Crane Manufacturers Association of America

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

NOTES

	HI	=	Hydraulic Institute
	API	=	American Petroleum Institute
	ANSI	=	American National Standards Institute
	ACI	=	American Concrete Institute
	IEEE	=	Institute of Electrical and Electronic Engineers
	AISC	=	American Institute of Steel Construction
	NA	=	Not Applicable
	TEMA	=	Tubular Exchanger Manufacturer's Association
3.	I	=	Designed in accordance with the seismic requirements of Seismic Category I structures and equipment as described in Section 3.7, "Seismic Design".
	a	=	Equipment meets the intent of the seismic design requirements of Branch Technical Position (ETSB 11-1 (Rev. 1) where applicable.
	NA	=	Seismic Category I requirements are not applicable to the structure or equipment.
4.	B	=	Equipment meets the QA requirements of 10CFR50, Appendix B, and the QA Program Description (QAPD).
	-	=	The QA requirements of 10CFR50, Appendix B, are not mandatory.
	a	=	Equipment meets the intent of the QA requirements of Branch Technical Position ETSB 11-1 (Rev.1) B.VI, "Quality Assurance for Radioactive Waste Management Systems" (November 24, 1975).
	b	=	Equipment meets the QA requirements of Branch Technical Position ASB 9.5-1, IV.B.7, "Quality Assurance Program".
	c	=	Coatings meet the intent of RG 1.54.

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

NOTES

d	=	In accordance with RG 1.29, Regulatory Position C.4, the pertinent provisions of 10CFR50, Appendix B, are applied during the operations phase.
e	=	The pertinent provisions of 10CFR50, Appendix B, are applied during the construction phase.
f	=	This equipment is not safety-related. The meteorological data collection programs which control these activities are subject to the pertinent provisions of 10CFR50 Appendix B which are described in the Operations Quality Assurance Plan.
NA	=	The QA requirements of 10CFR50, Appendix B, are not applicable.

5. All vent and drain piping and valves are classified NNS after the isolation valve, and are designed to ANSI B31.1.

6. During and after a seismic event the component and its supports are designed to retain structural integrity and prevent collapse and damage to safety-related equipment and structures. Operability need not be retained. Also see note 17.

7. Not used.

8. Actual Q-valves, dampers, strainers and lines are identified on the system P&IDs, system isometric drawings, and/or piping class summary sheets and Instrument Index.

9. Supports of components with a nuclear safety function are the same safety class as the components they support.

10. Equipment is not safety-related but is required to function during and after a SSE or OBE, as applicable. See RG 1.45 for RCPB instrumentation.

11. Table indicates the required code and/or seismic category based on its safety-related importance as dictated by service and functional requirements and the consequences of failure. The actual equipment may be designed to code, quality assurance, and/or seismic guidelines which are higher than required.

TABLE 3.2.A-1 (Continued)

BALANCE OF PLANT-QUALITY CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

NOTES

12. Depending on the qualification category of the equipment as defined in Appendix 7A and in RG 1.97, the following QA requirements apply:

    - Category 1 equipment meets the QA requirements of 10CFR50, Appendix B.

    - Category 2 equipment meets a modified program similar to that program for fire protection and radioactive waste management equipment.

    - No special QA requirements are applicable for Category 3 equipment.

13. The actual code class break is extended to the first weld outside the isolation valve cubicle north wall for support and operability reasons.

14. The radiation monitoring system components are discussed in Section 11.5 and 12.3, and include both the safety-related Class 1E monitors and RG 1.97, Category 2, monitors.

15. Not used.

16. Not used.

17. HVAC ductwork and dampers, electrical cable trays and conduits may be attached to Non-Nuclear Safety (NNS) steel angle face frames provided around penetrations through concrete walls or slabs.

    The design practice at STPEGS did not rely on the specified minimum yield strength of angles or on the welded fabrication of the angle frames. Accordingly, the control of these attributes through the QA requirements of a project nuclear safety classification was not required to provide adequate support for the designed commodities. All angle frames installed in Category I structures after April 24, 1987, will meet safety-related QA program requirements.

18. The Fuel Building Cask Handling Crane bridge is designed per CMAA 70. The trolley is designed to NUREG-0554 and ASME NOG-1. The overall crane complies with NUREG-0612. CN-3133

TABLE 3.2.A-2

CORRELATION OF SAFETY CLASSIFICATION
WITH INDUSTRY CODES AND STANDARDS FOR MECHANICAL COMPONENTS [(a)] [(b)]

Components	Safety Class 1	Safety Class 2	Safety Class 3	NNS
Pressure vessels and HXs	ASME Section III Class 1	ASME Section III Class 2	ASME Section III Class 3	ASME Section VIII Division 1
Pumps	ASME Section III Class 1	ASME Section III Class 2	ASME Section III Class 3	ASME Section VIII Division 1 (see note c) & Manufacturer's Standard
Piping and valves	ASME Section III Class 1	ASME Section III Class 2	ASME Section III Class 3	ANSI B31.1 Power Piping
Metal Containment components	-	ASME Section III Class MC	-	-
Atmospheric storage tanks (Note d)	-	ASME Section III Class 2	ASME Section III Class 3	API-650, AWWA-D100, ANSI B96.1 or equivalent
0-15 psig storage tanks (Note d)	-	ASME Section III Class 2	ASME Section III Class 3	API-620 or equivalent

a. With options and additions as necessary for service conditions and environmental requirements.
b. Components of the RCPB shall meet the requirements of 10CFR50, Section 50.55a, "Codes and Standards." All other components shall satisfy codes and addenda in effect at the time of component order.
c. For pumps classified NNS and operating above 150 psi or 212°F, ASME Code, Section VIII, Division 1 shall be used as a guide in calculating the wall thickness for pressure-retaining parts and in sizing the cover bolting. For pumps operating below 150 psi and 212°F, manufacturer's standard pump for service intended may be used. No code stamping is required.
d. These codes and standards do not apply to concrete tanks.

TABLE 3.2.B-1

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE FLUID SYSTEM COMPONENTS*

System or Component	Safety Class[1]	Standard or Code[12]	Code Class	Quality Assurance	Seismic Design[13]	Remarks
Reactor Coolant System						
Reactor vessel	1	ASME III	1	Note 3	I	
CRDM housing	1	ASME III	1	Note 6	I	
Steam generator (tube side)	1	ASME III	1	Note 6	I	Note 21
(shell side)	2	ASME III	2	Note 6	I	
Pressurizer	1	ASME III	1	Note 6	I	
Reactor coolant hot and cold leg piping, supports, fittings and fabrication	1	ASME III	1	Note 3	I	For SCs of other piping and associated valves in the Reactor Coolant System and other auxiliary systems, see Note 7
Reactor vessel head vent system piping valves and supports	1, 2, NNS	ASME III ANSI B31.1	1, 2	Note 3	I	

* See Notes at end of Table 3.2.B-2 for abbreviations.

TABLE 3.2.B-1 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE FLUID SYSTEM COMPONENTS*

System or Component	Safety Class(1)	Standard or Code(12)	Code Class	Quality Assurance	Seismic Design(13)	Remarks
Reactor Coolant System (Continued)						
Surge pipe, supports, fittings, and fabrication	1	ASME III	1	Note 3	I	
Crossover leg piping, supports, fittings, and fabrication	1	ASME III	1	Note 3	I	
Pressurizer safety valves	1	ASME III	1	Note 3	I	
Pressurizer power-operated relief valves	1	ASME III	1	Note 3	I	
Pressurizer PORV block valves	1	ASME III	1	Note 3	I	
Other valves	1, 2, 3, NNS	ASME III, ANSI B31.1	1, 2, 3 NA	Note 25, NA	I, NA	Note 7
Pressurizer relief tank	NNS	ASME VIII		NA	NA	Note 23
Reactor coolant pump						
RCP casing	1	ASME III	1	Note 6	I	
Main flange	1	ASME III	1	Note 6	I	
Thermal barrier	1	ASME III	1	Note 6	I	
Thermal barrier HX	1	ASME III	1	Note 6	I	
Seal housing #1	1	ASME III	1	Note 6	I	
#2	2	ASME III	2	Note 6	I	Notes 8a and 21
Pressure-retaining bolting	1	ASME III	1	Note 6	I	

TABLE 3.2.B-1 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE FLUID SYSTEM COMPONENTS*

System or Component	Safety Class(1)	Standard or Code(12)	Code Class	Quality Assurance	Seismic Design(13)	Remarks
Reactor Coolant System (Continued)						
RCP motor	2	NEMA MG1		Note 6	I	
Motor rotor	2	Note 9 and 30		Note 6 and 30	I	
Motor shaft	2	Note 9 and 29		Note 6 and 29	I	
Shaft coupling	2	Note 9		Note 6	I	
Spool piece	2	Note 9		Note 6	I	
Flywheel	2	Note 9		Note 6	I	
Bearing (motor upper thrust)	2	Note 9		Note 6	I	
Motor bolting	2	Note 9		Note 6	I	Applies only to bolting involved with coastdown function
Motor stand	2	Note 9		Note 6	I	
Motor frame	2	Note 9		Note 6	I	
Upper oil reservoir (UOR)	3	No Code		Note 6	I	
UOR cooling coil	3	ASME III	3	Note 6	I	
Lower oil reservoir (LOR)	3	No Code		Note 6	I	
LOR cooling coil	3	ASME III	3	Note 6	I	
Oil cooler piping	3	No Code		Note 6	I	
Motor air coolers	3	ASME III	3	Note 6	I	Note 2

TABLE 3.2.B-1 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE FLUID SYSTEM COMPONENTS*

System or Component	Safety Class(1)	Standard or Code(12)	Code Class	Quality Assurance	Seismic Design(13)	Remarks
Chemical & Volume Control System						
Regenerative HX	2	ASME III	2	Note 3	I	
Letdown HX (tube side)	2	ASME III	2	Note 3	I	
(shell side)	3	ASME III	3	Note 3	I	
Mixed-bed demineralizer	3	ASME III	3	Note 4	NA(11)	
Cation-bed demineralizer	3	ASME III	3	Note 4	NA(11)	
Reactor coolant filter	2	ASME III	2	Note 3	I	
Volume control tank	2	ASME III	2	Note 3	I	
Centrifugal charging pump	2	ASME III	2	Note 3	I	
Positive displacement pump	2	ASME III	2	Note 3	I	
Seal water injection filter	2	ASME III	2	Note 3	I	
Letdown orifices	2	ASME III	2	Note 4	I	
Excess letdown HX						
(tube side)	2	ASME III	2	Note 3	I	
(shell side)	2	ASME III	2	Note 3	I	
Seal water return filter	2	ASME III	2	Note 3	I	
Seal water HX (tube side)	2	ASME III	2	Note 3	I	
(shell side)	3	ASME III	3	Note 3	I	Note 2
Boric acid transfer pump	3	ASME III	3	Note 3	I	

TABLE 3.2.B-1 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE FLUID SYSTEM COMPONENTS

System or Component	Safety Class[1]	Standard or Code[12]	Code Class	Quality Assurance	Seismic Design[13]	Remarks
Chemical & Volume Control System (Cont'd)						
Boric acid filter	3	ASME III	3	Note 4	I	
Boric acid batching tank	NNS	ASME VIII		NA	NA	Note 23
Chemical mixing tank	NNS	ASME VIII		NA	NA	Note 5
Concentrated boric acid sample cooler	NNS	ASME VIII		NA	NA	Notes 21 and 22
Concentrated boric acid polishing demineralizer	NNS	ASME VIII		NA	NA	Notes 21 and 22
Concentrated boric acid polishing filter	NNS	ASME VIII		NA	NA	Notes 21 and 22
Boron meter	NNS	ANSI B31.1		NA	NA	Classified on basis that flow restriction is provided in the piping
RC purification pump	3	ASME III	3	Note 4	I	
RCP standpipe and supports	NNS	ASME III		NA	NA	Note 5
Valves	1, 2, 3, NNS	ASME III ANSI B31.1	1, 2, 3 NA	Note 25, NA	I, NA	Note 7

TABLE 3.2.B-1 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE FLUID SYSTEM COMPONENTS

System or Component	Safety Class[1]	Standard or Code[12]	Code Class	Quality Assurance	Seismic Design[13]	Remarks
Boron Thermal Regeneration Subsystem						
Moderating HX	NNS	ASME VIII		NA	NA	Notes 21 and 23
Letdown chiller HX	NNS	ASME VIII		NA	NA	Notes 21 and 23
Letdown reheat HX						
(tube side)	2	ASME III		Note 3	I	
(shell side)	NNS	ASME VIII		NA	NA	Notes 21 and 22
Thermal regeneration demineralizer	NNS	ASME VIII		NA	NA	Notes 21 and 23
Chiller pump	NNS	No Code		NA	NA	Note 23
Chiller surge tank	NNS	ASME VIII		NA	NA	Note 5
Chiller unit	NNS	ASME VIII		NA	NA	Note 23
Evaporator	NNS	ASME VIII		NA	NA	Note 23
Condenser	NNS	ASME VIII		NA	NA	Note 23
Compressor	NNS	No Code		NA	NA	Note 23
Valves (W supplied)	NNS	ANSI B31.1		NA	NA	
Emergency Core Cooling System (Safety Injection System)						
Accumulator	2	ASME III	2	Note 3	I	
HHSI pump	2	ASME III	2	Note 3	I	
LHSI pump	2	ASME III	2	Note 3	I	
Valves (W supplied)	1, 2 NNS	ASME III ANSI B31.1	1, 2 NA	Note 25 NA	I, NA	Note 7

TABLE 3.2.B-1 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE FLUID SYSTEM COMPONENTS

System or Component	Safety Class(1)	Standard or Code(12)	Code Class	Quality Assurance	Seismic Design(13)	Remarks
Residual Heat Removal System						
Residual heat removal pump	2	ASME III	2	Note 3	I	
Residual HX (tube side)	2	ASME III	2	Note 3	I	
(shell side)	3	ASME III	3	Note 3	I	
Valves (W supplied)	1, 2 NNS	ASME III ANSI B31.1	1, 2 NA	Note 25 NA	I, NA	Note 7
Containment Spray System						
Containment spray pump	2	ASME III	2	Note 3	I	
Spray additive eductor	2	ASME III	2	Note 4	I	
Containment spray nozzles	2	ASME III	2	Note 4	I	
Valves						
a. Required for initial injection or long run recirculation of sump water	2	ASME III	2	Note 3	I	
b. Required for chemical addition	3	ASME III	3	Note 3	I	
c. Operators for safety related valves	NA	IEEE 279, 323	-	Note 3	I	

TABLE 3.2.B-1 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE FLUID SYSTEM COMPONENTS

System or Component	Safety Class(1)	Standard or Code(12)	Code Class	Quality Assurance	Seismic Design(13)	Remarks
Spent fuel Pool Cooling and Cleanup System [1,2]						
Spent fuel pool HX	3	ASME III	3	Note 3	I	
Spent fuel pool cooling pump	3	ASME III	3	Note 3	I	
Spent fuel pool filter	NNS	ASME VIII		NA	NA	Notes 10 and 23
Spent fuel pool demineralizer	NNS	ASME VIII		NA	NA	Notes 10 and 23
Spent fuel pool strainer	NNS	No Code		NA	NA	Note 5
Spent fuel pool skimmer pump	NNS	MS		NA	NA	Notes 21 and 22
Spent fuel pool skimmer suction head	NNS	No Code		NA	NA	Note 23
Spent fuel pool skimmer filter	NNS	ASME VIII		NA	NA	Notes 10 and 23
Refueling water purification pump	NNS	ASME III		NA	NA	Note 21
Valves	2, 3, NNS	ASME III, ANSI B31.1	2, 3, NA	Note 25, NA	I, NA	Note 7

TABLE 3.2.B-1 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE FLUID SYSTEM COMPONENTS

System or Component	Safety Class(1)	Standard or Code(12)	Code Class	Quality Assurance	Seismic Design(13)	Remarks
Boron Recycle System						
Recycle evaporator feed pump	NNS	MS		NA	NA	Notes 21 and 23
Recycle evaporator feed demineralizer	NNS	ASME VIII		NA	NA	Notes 21 and 23
Recycle evaporator feed filter	NNS	ASME VIII		NA	NA	Notes 21 and 23
Recycle evaporator condensate demineralizer	NNS	ASME VIII		NA	NA	Note 23
Recycle evaporator condensate filter	NNS	ASME VIII		NA	NA	Note 23
Recycle evaporator concentrate filter	NNS	ASME VIII		NA	NA	Note 23
Recycle evaporator reagent tank	NNS	ASME VIII		NA	NA	Note 5
Recycle evaporator package						
1. Feed preheater	NNS	ASME VIII		NA	NA	Notes 21 and 23
2. Gas stripper	NNS	ASME VIII		NA	NA	Notes 21 and 23
3. Submerged tube evaporator	NNS	ASME VIII		NA	NA	Notes 21 and 23
4. Evaporator condenser	NNS	ASME VIII		NA	NA	Notes 21 and 23
5. Distillate cooler	NNS	ASME VIII		NA	NA	Notes 21 and 23
6. Absorption tower	NNS	ASME VIII		NA	NA	Notes 21 and 23
7. Vent condenser	NNS	ASME VIII		NA	NA	Notes 21 and 23

TABLE 3.2.B-1 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE FLUID SYSTEM COMPONENTS

System or Component	Safety Class[1]	Standard or Code[12]	Code Class	Quality Assurance	Seismic Design[13]	Remarks
Boron Recycle System (Cont'd)						
8. Distillate pump	NNS	MS		NA	NA	Notes 21 and 23
9. Distillate condenser	NNS	ASME VIII		NA	NA	Notes 21 and 23
10. Concentrate pump	NNS	MS		NA	NA	Notes 21 and 23
11. Piping and supports						
a. Feed	NNS	ANSI B31.1		NA	NA	Notes 21 and 23
b. Distillate	NNS	ANSI B31.1		NA	NA	Notes 21 and 23
c. Concentrate	NNS	ANSI B31.1		NA	NA	Notes 21 and 23
12. Valves	NNS					
a. Feed	NNS	ANSI B31.1		NA	NA	Notes 21 and 23
b. Distillate	NNS	ANSI B31.1		NA	NA	Notes 21 and 23
c. Concentrate	NNS	ANSI B31.1		NA	NA	Notes 21 and 23
d. Cooling	NNS	ANSI B31.1		NA	NA	Notes 21 and 23
e. Steam		ANSI B31.1		NA	NA	
Valves	NNS	ANSI B31.1		NA	NA	
Main Steam System						
Main steam isolation valves	2	ASME III	2	Note 3	I	
Steam dump valves	NNS	ANSI B31.1		NA	NA	Notes 21 and 22
Main Feedwater System						
Feedwater control valves and FW bypass control valves	NNS	ANSI B31.1		NA	NA	Notes 21 and 22

TABLE 3.2.B-2

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE SUPPLIED NON-FLUID SYSTEM COMPONENTS *

	Safety Class(1)	Code(12)	Code Class	Quality Assurance	Seismic Design(13)	Remarks
Fuel-Handling System						
Semiautomatic refueling machine	NNS			NA	NA	Notes 16 and 23
Integrated head package 1. Parts providing seismic support to CRDMs, including	1	ASME III	NF-1	Note 3	I	See Note 19
missile shield and head lift rods. 2. Remainder of package	NNS			NA	NA	See Note 23
CRDM seismic support tie rods	1	ASME III	NF-1	Note 3	I	See Note 19
Fuel-handling machine	3			Note 4	I	Note 17
Rod cluster control changing fixture	NNS			NA	NA	Notes 16 and 23
Reactor vessel stud tensioner	NNS			NA	NA	Notes 5 and 15
Spent fuel assembly handling tool and telescoping fuel handling tool	NNS			NA	NA	Notes 5 and 10
Fuel Transfer System fuel transfer tube and flange	2	AMSE III	MC	Note 4	I	Portions of Containment boundary
Remainder of system	NNS			NA	NA	Notes 10 and 23

* See Note at end of table for abbreviations

TABLE 3.2.B-2 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE SUPPLIED NON-FLUID SYSTEM COMPONENTS

	Safety Class(1)	Code(12)	Code Class	Quality Assurance	Seismic Design(13)	Remarks
Fuel-Handling System (Cont'd)						
New fuel elevator	NNS			NA	NA	Notes 10 and 23
New fuel racks	3	ASME III	3	Note 6	I	Note 14
Incontainment fuel racks	3	ASME III	3	Note 6	I	Note 14
Portable underwater lights	NNS			NA	NA	Notes 5 and 10
Load cell	NNS			NA	NA	Monitors lifting of internals Note 23
Lower internals storage stand	NNS			NA	NA	Notes 10 and 23
Upper internals storage stand	NNS			NA	NA	Notes 10 and 23
Thimble plug handling tool	NNS			NA	NA	Notes 5 and 10
Primary source installation guide	NNS			NA	NA	Notes 5 and 10
Crane scales (3)	NNS			NA	NA	Notes 5 and 10
Stud tensioner handling devices (3)	NNS			NA	NA	Notes 5 and 10
Irradiation tube end plug seating jack	NNS			NA	NA	Notes 5 and 10
Burnable absorber rod assembly handling tool	NNS			NA	NA	Notes 5 and 10

TABLE 3.2.B-2 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE SUPPLIED NON-FLUID SYSTEM COMPONENTS

	Safety Class(1)	Code(12)	Code Class	Quality Assurance	Seismic Design(13)	Remarks
Fuel-Handling System (Cont'd)						
Irradiation sample handling tool	NNS			NA	NA	Notes 5 and 10
Burnable absorber rod assembly fuel rack inserts (26)	NNS			NA	NA	Notes 5 and 10
Control rod drive shaft handling fixture	NNS			NA	NA	Notes 10 and 23
Control rod drive shaft unlatching tool	NNS			NA	NA	Notes 5 and 10
New fuel elevator winch	NNS			NA	NA	Notes 10 and 23
Neutron detector positioners						
1. Parts that support safety related detectors	2			Note 4	I	
2. Remainder of system	NNS			NA	NA	
Reactor-vessel-to-refueling cavity seal ring	NNS			NA	NA	Any reasonably postulated failure would cause only operational inconvenience Note 5
New fuel assembly handling tool	NNS			NA	NA	Notes 10 and 23
New rod cluster control handling tool	NNS			NA	NA	Notes 5 and 10

TABLE 3.2.B-2 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE SUPPLIED NON-FLUID SYSTEM COMPONENTS

	Safety Class(1)	Code(12)	Code Class	Quality Assurance	Seismic Design(13)	Remarks
Fuel-Handling System (Cont'd)						
Reactor vessel internals lifting rig and o-ring change fixture	NNS			NA	NA	Note 23 and 27
o-ring retaining fixture	NNS			NA	NA	Notes 5 and 10
o-ring lifting device	NNS			NA	NA	Notes 5 and 10
Reactor Vessel or Core-Related						
Reactor vessel support shoes and shims	1			Note 4	I	Provides mechanical support for SC 1 component
Irradiation sample holder	2			Note 4	I	Note 18
Irradiation samples	NNS			NA	NA	Changed characteristics must be assessed later to ensure continued safe operation of the reactor vessel Note 22
Burnable absorber rod assemblies	NNS			NA	NA	Requires multiple failures to cause redistribution Note 22
Reactor vessel insulation	NNS			NA	NA	Notes 10 and 23

TABLE 3.2.B-2 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE SUPPLIED NON-FLUID SYSTEM COMPONENTS

	Safety Class[1]	Code[12]	Code Class	Quality Assurance	Seismic Design[13]	Remarks
Reactor Vessel or Core-Related (Cont'd)						
Reactor vessel internals	2			Note 6	I	The major internals direct flow, ensure core cooling, and prevent displacement of the core; other internals are SC 2 for reasons cited in Note 18
Full-length control rods	2			Note 3	I	Required to shut down core
CRDM dummy can assemblies	NNS			NA	NA	Notes 5 and 10
Primary source rods	NNS	-		NA	NA	Note 22
Fuel assemblies	2	-		Note 24	I	
Incore Instrumentation						
Seal table assembly	1	ASME III	1	Note 3	I	Provides support to the SC 1 pressure boundary conduit
Flux thimble tubing	2	ASME III	2	Note 3	I	Notes 20 and 28
Flux thimble fittings	2	ASME III	2	Note 3	I	Notes 20 and 28
Flux thimble guide tubing	1	ASME III	1	Note 3	I	Note 28
Instrumentation and Control System Components						
Reactor trip system	NA	IEEE 279, 323, 344		Note 3		This is a general classification of equipment.

TABLE 3.2.B-2 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE SUPPLIED NON-FLUID SYSTEM COMPONENTS

	Safety Class[1]	Code[12]	Code Class	Quality Assurance	Seismic Design[13]	Remarks
Instrumentation and Control System Components (Cont'd)						
ESF actuation system	NA	IEEE 279, 323, 344		Note 3	I	

TABLE 3.2.B-2

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE SUPPLIED NON-FLUID SYSTEM COMPONENTS

NOTES

1. 1, 2, 3, NNS = Safety classes defined in Section 3.2.B.2

   NA = Not applicable

2. Portions of equipment containing component cooling water are SC 3, Code Class 3.

3. Meets "Quality Control System Requirements," Westinghouse QCS-1, which satisfies requirements of 10CFR50, Appendix B, Quality Assurance Criteria. The operations QA program as described in the Operations Quality Assurance Plan is applicable.

4. Meets "Quality Requirements for Manufacture of Nuclear Plant Equipment," Westinghouse QCS-2, which satisfies requirements of 10CFR50, Appendix B. The operations QA Program as described in the Operations Quality Assurance Plan is applicable.

5. Access for inspection and test required by Westinghouse; however, no formal quality program approval required.

6. Meets the quality assurance program of one of the Westinghouse NES manufacturing divisions and/or subvendors, and is in accordance with 10CFR50, Appendix B. The operations QA Program as described in the Operations Quality Assurance Plan is applicable.

7. Safety classes for piping and valves are as defined by the P&IDs. Code classes are those required by the safety class.

8. Represents code class upgrading:

   8a. As permitted by paragraph NA-2134 of the ASME B&PV Code, Section III, this component is upgraded from the minimum required Code Class 2 to Code Class 1.

   8b. As permitted by paragraph NA-2134 of the ASME B&PV Code, Section III, this component is upgraded from the minimum required Code Class 3 to Code Class 2.

9. Parts are mechanically of safety class and must meet the structural integrity requirements of the specification and quality assurance requirements of 10CFR50, Appendix B.

10. Failure can cause no nuclear safety problem, although an economic loss may result.

TABLE 3.2.B-2 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE SUPPLIED NON-FLUID SYSTEM COMPONENTS

NOTES

11. This component is SC 3 under the definition 2.2.3 (1), (3), or (4) of ANSI N18.2-1973 and qualifies for no special seismic design by meeting the four conditions listed in Section 3.2.B.1. Portions of systems in which this component is located that perform the same safety function likewise qualify for no special seismic design.

12. ASME: American Society of Mechanical Engineers. III stands for Section III of the ASME Boiler and Pressure Vessel (B&PV) Code. VIII stands for Section VIII of the ASME B&PV Code. The earliest applicable Code for the pressure vessels which are part of the RCPB is the 1971 edition with application of all addenda through, to, and including the winter 1972 addenda. The earliest applicable Code for the pumps and valves which are part of the RCPB is 1971 edition, with application of all addenda through, to, and including the winter 1972 addenda. The earliest applicable Code for the piping which is part of the RCPB is the 1971 edition, with application of all addenda through, to, and including the summer 1973 addenda. Later codes may be used optionally (see Table 5.2-1).

13. Information as to seismic qualification methods is given in Sections 3.7, 3.9 and 3.10.

14. Must maintain fuel array to prevent criticality under adverse conditions including occurrence of the Design Basis (Safe Shutdown) Earthquake.

15. To be safety classified, failure of the tool must be directly a nuclear safety problem. If a nuclear safety problem arises from tool failures combined with a procedural failure thereafter, the tool is NNS.

16. Failure occurs inside isolable Reactor Containment; substantial release to the environment of radioactive gases from damaged spent fuel is prevented by isolation.

17. Failure of equipment outside Reactor Containment could cause substantial releases of radioactive gases from damaged spent fuel.

18. Any reactor vessel internal, the single failure of which could cause release of mechanical piece having potential for direct damage (as to the vessel cladding) or flow blockage, shall be classified to a minimum of SC 2.

19. These items are required as mechanical supports for CRDM housings during OBE and SSE.

20. Failure could cause a Loss-of-Coolant Accident, but less than a Condition III loss of coolant.

TABLE 3.2.B-2 (Continued)

EQUIPMENT CODE AND CLASSIFICATION LIST
WESTINGHOUSE SUPPLIED NON-FLUID SYSTEM COMPONENTS

NOTES

21. The table indicates the required code and seismic categories based on safety-related importance as dictated by service and functional requirements and by the consequences of their failure. These components and piping may have been designed to code, quality assurance, and/or seismic guidelines which are higher than required.

22. Equipment meets "Quality Control System Requirements" Westinghouse QCS-1; however, no quality assurance program is required.

23. Equipment meets "Quality Requirements for Manufacture of Nuclear Power Plant Equipment," Westinghouse QCS-2; however, no quality assurance program is required. The Operations QA Program as described in the Operations Quality Assurance Plan is applicable.

24. Equipment meets QA program outlined in Westinghouse Quality Management System. The Operations QA program as described in the Operations Quality Assurance Plan is applicable.

25. Quality Assurance programs for safety class valves meet the requirements of 10CFR50, Appendix B as appropriate.

26. The Fuel Handling System conveyer system (FHB side) was originally built to SC 3 and seismic Category I standards.

27. The Reactor Vessel Internals Lifting Rig and O-Ring Change fixture is not designed to seismic Category I requirements, but it is seismically stored on the O-Ring Retaining Fixture (located inside of the seismically designed Reactor Head Storage Stand) during normal operation. While it is in this storage position, seismic restraints are attached to prevent damage to adjacent equipment.

28. The ASME III requirement is for the original installation. For replacements, the exemptions provided by ASME XI Section IWA-4000 may be used.

29. The RCP Motor shaft-to-flywheel and shaft-to-thrust-runner fits may be restored to specification using non-safety-related build-up or plating process, provided that 1) a safety-related machining process is used and controlled to ensure no more than 0.020" (for shaft-to-flywheel fit) or 0.015" (for shaft-to-thrust-runner fit) is removed from the shaft diameter prior to plating or build-up and 2) a safety-related process is used to verify that final measurements meet specifications.

30. Non-safety-related tack welding of rotor laminations may be performed to stabilize the position of rotor laminations.

TABLE 3.2.C-1

GENERIC LETTER 89-09 REPLACEMENT LIST

Vendor	Item Description	Purchase Order	Part Number (Note 1)	Intended/Potential Applications
Cooper Industries, Inc.	Ball, Valve, 3 in., 150 lb. (Note 2)	RS25444	581-4089	Outside Containment Isolation Valve Reactor Makeup Water Supply to RCB TPNS B1RCFFV3651 & B2RCFV3651
Oil Tool Div. (WKM Valve)	Ball, Valve, 3 in., 150 lb.	QS7188	581-4089	Outside Containment Isolation Valve Reactor Makeup Water Supply to RCB TPNS B1RCFV3651 & B2RCFV3651
Hayward Tyler, Inc.	Cover, Back	RS8655	583-2459	Reactor Makeup Water Pump TPNS 3R271NPA101A & B, 3R272NPA201A & B
	Casing Casting	RS8655	583-2434	Reactor Makeup Water Pump TPNS 3R271NPA101A & B, 3R272NPA201A & B

NOTES

1. Part number provided for information only.

2. Ball is material only, not fabricated by welding. Materials were procured by the Station from QSC holder. Ball machined to proprietary dimensions by Cooper.

3.3 WIND AND TORNADO LOADINGS

The design bases for the safety-related structures, systems and components are outlined in General Design Criteria (GDC) 2, “Design Bases for Protection Against Natural Phenomena,” of Appendix A 10CFR50.

Structures, systems and components that perform a safety function are protected from failure due to tornado and wind loadings or missiles because they are either designed to withstand wind and tornado effects or are housed within a structure that is designed to withstand wind and tornado effects.

3.3.1 Wind Loadings

The procedures outlined below are based on “Building Code Requirements for Minimum Design Loads in Buildings and Other Structures, “American Nuclear Standard Institute (ANSI) A58.1-1972, hereinafter referred to as the ANSI Code (Ref. 3.3-1).

3.3.1.1 Design Wind Velocity. As required by Regulatory Guide (RG) 1.70, a design wind velocity based on the fastest mile wind speed, 30 ft above ground, 100-year mean recurrence interval has been selected. The design wind velocity selected for South Texas Project Electric Generating Station (STPEGS) is 125 mph. However, it should be noted that the design tornado (see Section 3.3.2.1) parameters include winds with a tangential velocity of 290 mph and a translational velocity of 70 mph (maximum). For design calculations, the tornado wind loading is assumed to be 360 mph, almost three times the design wind velocity. Since the tornado-generated winds and resulting forces are greater, the design tornado parameters govern the design of Category I structures.

3.3.1.1.1 Basis for Design Wind Velocity Selection: The design wind velocity for STPEGS is selected based on information in the ANSI Code, and on calculated faster mile speeds using empirical evidence. The design wind velocity as shown on the ANSI map “Annual Extreme Fastest-Mile Speed 30 Feet Above Ground, 100-Year Mean Recurrence Interval,” is between 90 and 100 mph. The design wind velocity was chosen to be 125 mph for additional conservatism.

3.3.1.1.2 Vertical Velocity Distribution and Gust Factors: In Subsection 3.1.1.1, wind is defined by its basic design velocity, i.e., as a perfectly smooth, laminar motion of air at a constant speed.

To account for discrepancies between the above model and nature, the following corrections are made:

1. Variation of wind velocity with height is compensated for by the introduction of velocity distribution coefficients, as indicated by the following expression.

$$V_Z = V_{30}\left(\frac{Z}{30}\right)^{y}$$

where:

$\left(\frac{Z}{30}\right)^{y}$ = Velocity distribution coefficient

V_Z = Wind Velocity at Z ft above ground (mph)

where:

V_{30} = Wind Velocity at 30 ft above ground (mph)

y = 0.143 = Velocity distribution factor

Z = Height above ground in ft

The value of velocity distribution factor, y, is the same as that used by ANSI Code for the exposure C (flat, open country, open flat coastal belts, and grasslands).

2. Sudden brief fluctuations in the wind speed (gusts) and the dynamic nature of load are accounted for through application of the gust factors.

The gust factors, G_f (for buildings and structures) and G_p (for parts and portions), are assigned the same values as those suggested by the ANSI Code. They provide conservatively for the dynamic response of ordinary buildings. In cases where the ratio of building height to the least horizontal dimension exceeds 5, a detailed analysis of building dynamic response is performed by using the method described in Section A.6.3.4.1 of the ANSI Code.

3.3.1.2 Determination of Applied Forces.

3.3.1.2.1 Effective Velocity Pressures: Design wind velocities are converted into pressures by means of the following expressions.

$$q_F = 0.00256\ V_{30}^2\ K_Z\ G_f$$

$$q_p = 0.00256\ V_{30}^2\ K_Z\ G_p$$

where:

q_f = Effective velocity pressure for buildings and structures, lb/ft^2

q_p = Effective velocity pressure for parts and portions of buildings and structures, lb/ft^2

0.00256 = $\frac{1}{2}$ (mass density of air) (velocity conversion factor)

$$= \frac{1}{2}\left(\frac{.07651}{32.2}\right)\left(\frac{5280}{3600}\right)^2$$

K_Z = Velocity pressure coefficient for height and exposure, $K_Z = (V_Z/V_{30})^2$

G_f, G_p, V_{30} are as previously defined.

The effective velocity pressures for various heights above ground are shown in Table 3.3-1.

3.3.1.2.2 Design Wind Pressures: Wind forces on a structure, or any element thereof, result from a differential pressure caused by the obstruction of the free flow of the wind.

Therefore, in addition to being proportional to wind velocity, the design wind pressures are a function of the orientation, shape and size of the object obstructing the free flow of wind. They are obtained by multiplying the effective velocity pressure by the pressure coefficients given in Table 3.3-2, as indicated by the following expressions:

$$P_{jF} = C_p q_F + C_{pi} q_M$$

$$P_{jp} = (C_p \text{ or } C_{pl})\ q_p + C_{pi}\ q_M$$

where:

P_{jF} = Design wind pressure for buildings and structures, lb/ft^2

P_{jp} = Design wind pressure for parts and portions, lb/ft^2

C_p = External pressure coefficient

C_{pl} = External local pressure coefficient

Pressure q_pC_{pl} is used for the corners of all walls, and the ridges, eaves, cornices and 90-degree corners of roofs. For walls, the pressure is assumed to act over vertical strips of width 0.10 w, where w is the least width of the building. For roofs, the pressure is assumed to act over strips of width 0.10 d, where d is the least width of the building normal to ridge. Local pressures q_pC_{pl} are applied outward.

C_{pi} = Internal pressure coefficient

q_F = Effective velocity pressure for buildings and structures, lb/ft^2

q_M = Effective velocity pressure for internal pressure calculations, lb/ft^2

q_p = Effective velocity pressure for parts and portions, lb/ft^2

3.3.1.2.3 Design Wind Loads: The design wind loads, W_F and W_p are defined by:

$$W_F = \sum_{j=1}^{n} P_{jF} A_{jF}$$

$$W_p = \sum_{j=1}^{n} P_{jp} A_{jp}$$

where:

W_F = Design wind load (lb) for buildings and structures

W_p = Design wind load (lb) for parts and portions of buildings or structures

A_{jp} & A_{jf} = Exposed areas, ft^2

P_{jp} & P_{jf} = As defined in Section 3.3.1.2.2

j & n = Summation indices specifying that summation takes place over all exposed areas

3.3.2 Tornado Loadings

This subsection describes provisions for calculating tornado-generated forces on structures and parts and portions thereof.

3.3.2.1 Applicable Design Parameters. The design parameters have been selected based on tornado observations reported in technical literature, and on commonly accepted engineering practice. Category I structures are designed to withstand effects of a tornado having the following characteristics: (Ref. 3.3-5).

Translational Velocity	70 mph (maximum) 5 mph (minimum)
Tangential Velocity	290 mph
Atmospheric Pressure Drop	3 psi
Rate of Pressure Drop	2 psi/sec
Radius of Maximum Wind Speed	150 ft

Tornado-generated missiles are specified in Section 3.5.1.4

3.3.2.2 Determination of Forces on Structures. In the evaluation of tornado wind effects on Category I structures, systems, and components, the potential damage due to several phenomena has been considered. These phenomena are:

1. Velocity Pressures – forces resulting from a transfer of the kinetic energy of wind to a structure obstructing the free flow of air masses

2. Atmospheric Pressures Change – forces created by the rapid pressure changes in tornado vortex

3. Impact Forces – created by missiles generated by extreme windspeeds associated with a tornado

3.3.2.2.1 Velocity Pressures: The velocity pressure calculations are based on procedures outlined in Subsection 3.3.1.2 taking velocity distribution coefficients, K_Z, and gust factors, G_f and G_p equal to unity.

For application of velocity pressure on the Containment structure and auxiliary feedwater storage tank, References 3.3-2 and 3.3-3 are used in addition to ANSI A58.1-1972.

The effective velocity pressures are found using tornado parameters specified in Subsection 3.3.2.1, and the tangential velocity distribution given by the following expression:

$$V_T(r) = \begin{cases} C\left(\dfrac{r}{R_c}\right) & \text{for } r \leq R_c \\ C\left(\dfrac{R_c}{r}\right) & \text{for } r > R_c \end{cases}$$

where:

$V_T(r)$ = Tangential velocity at distance r from center of vortex (COV), mph

r = Radial distance from COV, ft

R_c = Radius of the maximum wind speed, ft

C = Constant = 290 mph

The values of effective velocity pressures are shown on Figure 3.3-1.

3.3.2.2.2 Atmospheric Pressure Drop: The circular pattern of air motion in a tornado produces an atmospheric pressure drop within the vortex. The pressure drop is a function of tangential wind velocity and distance from the center of vortex, and can be determined by making use of the cyclostropic wind equation, given in Reference 3.3-4:

$$\frac{1}{\rho}\,\frac{d\{\Delta p(r)\}}{dr} = \frac{\{V_T(r)\}^2}{r}$$

where:

Δp (r) represents atmospheric pressure drop, ρ is mass density of air.

V_T (r) and r are defined above.

The above equation was solved by a numerical method, and the values of pressure differential plotted on Figure 3.3-2.

For the design of exterior structural elements (walls, panels, roof slabs, etc.) no credit was taken for venting.

3.3.2.2.3 Tornado-Generated Missiles: Tornado-generated missile parameters are presented in Table 3.5-9.

3.3.2.2.4 Combination of Applied Loads: For each particular structure or portion thereof, the most adverse combination is obtained by placing the structure under consideration at various locations in the tornado field (at various distances from the COV) to determine the maximum local and overall effects on the structure resulting from the wind velocity pressure, and the atmospheric pressure drop by making use of Figures 3.3-1 through 3.3-3.

Once the effective loads for the individual tornado generated effects (i.e., velocity pressure, differential pressure and missile load) are established, the governing combination is obtained as the most adverse of the following:

$W_T = W_w$

$W_T = W_p$

$W_T = W_m$

$W_T = W_w + 0.5W_p$

$W_T = W_w + W_m$

$W_T = W_w + 0.5W_p + W_m$

where:

W_T = Total tornado load

W_w = Tornado wind velocity pressure load

W_p = Tornado differential pressure load

W_m = Tornado missile load.

For overall structural effects such as overturning, sliding and torsion, the design pressure applied to the exterior surface of a structure is the actual pressure calculated on a plane that passes through the center of the structure as shown on Figure 3.3-3.

For local structural effects, the maximum design pressure, is determined by combining the pressure drop, P, and the effective velocity pressure, q_F, and is used to design small building surfaces and is applied uniformly.

The effect of a tornado-generated missile is determined by transforming the impactive dynamic forces into effective loads (using energy balance methods) and combining these with the effects of the design pressure, as stated in total design tornado loads, W_T.

These total tornado loads have then been combined with the other loads to design structures as specified in Section 3.8.1, 3.8.4, and 3.8.5.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads. To ensure the ability of Category I structures to perform despite failure of structures not designed for tornado loads, the following criteria are met:

1. The plant arrangement provides for sufficient separation between Category I structures and non-Category I structures so that failure of the latter cannot affect the ability of Category I structures to perform their safety functions.

2. Where the above criteria are not met, the affected non-Category I structure has been designed either to withstand tornado loads or not to collapse against Category I structures under tornado loadings.

3. The tornado missile parameters considered in the design of Category I structures (see Section 3.5) encompass the spectrum of missiles which could be generated as a result of failure of structures or equipment not designed to withstand tornado loading.

The systems and components in safety-related structures are either protected from the effects of tornado by their enclosure, or are checked to ensure that the system or components can withstand depressurization or that their failure will not affect the ability of other structures, systems and components to perform their intended safety function or analyzed to demonstrate that the probability of site proximity missiles, adversely affecting safety-related structures, systems and components is acceptably low.

REFERENCE

Section 3.3:

3.3-1 ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures".

3.3-2 ASCE 3269, "Wind Forces on Structures", American Society of Civil Engineers, Transactions, Vol. 126, Part II, 1961.

3.3-3 Maher, F. J., "Wind Loads on Dome – Cylinder and Dome – Cone shapes", Journal of the Structural Division, ASCE Vol. 92, No. ST5, Proc. Paper 4933, October 1966.

3.3-4 Stevenson, J. D., Tornado Design of Class I Structures for Nuclear Power Plants, Westinghouse Nuclear Energy Systems, Pittsburgh, Pennsylvania.

3.3-5 NRC Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants", April 1974.

TABLE 3.3-1

EFFECTIVE VELOCITY PRESSURES

(v_{30} = 125 MPH)

Z (ft)	q_F (psf)	q_p (psf)	q_M (psf)
< 30	42	60	40
30	52	60	40
50	59	67	47
100	69	76	57
150	75	84	63
200	80	89	69
250	84	94	73
300	87	96	77
350	90	100	81
400	93	103	84
450	96	105	87
500	98	108	89
550	100	110	92
600	102	113	95
650	104	114	96
700	106	117	99
750	108	119	100
800	110	120	102

TABLE 3.3-2
PRESSURE COEFFICIENTS SELECTION GUIDE

RECTANGULAR BUILDINGS	EXTERNAL PRESSURE COEFFICIENTS	WALLS		GLOBAL	TABLE 7 in REF. 3.3-1
				LOCAL	SUBSECTION 6.5.3.1 in REF. 3.3-1
		ROOFS	ARCH	GLOBAL	WIND PARALLEL TO AXIS: SUBSECTION 6.5.3.2.1, REF. 3.3-1 WIND PERPENDICULAR TO AXIS: TABLE 8, REF. 3.3-1
				LOCAL	TABLE 10 & SUBSECTION 6.5.3.2.4 in REF. 3.3-1
			GABLED	GLOBAL	WIND PARALLEL TO RIDGE: SUBSECTION 6.5.3.2.1, REF. 3.3-1 WIND PERPENDICULAR TO RIDGE: SUBSECTION 6.5.3.2.3 & TABLE 9, REF. 3.3-1
				LOCAL	TABLE 10 & SUBSECTION 6.5.3.2.4 in REF. 3.3-1
	INTERNAL PRESSURE COEFFICIENTS				TABLE 11 & SECTION 6.5.4 in REF. 3.3-1
OTHER STRUCTURES	STACKS				TABLE 15 & SECTION 6.7 in REF. 3.3-1
	SPHERES				TABLE 7 in REF. 3.3-3
	CYLINDERS				TABLE 4 (f) in REF. 3.3-2

3.4 WATER LEVEL (FLOOD) DESIGN

The methods of analysis used to determine the design basis flood are discussed in Section 2.4. These methods are consistent with the requirements of Regulatory Guide (RG) 1.59.

The protection measures used to accommodate static and dynamic flood loads on Category I structures generally fall under the category of "incorporated barriers" as specified in regulatory position C.1 of RG 1.102.

3.4.1 Flood Protection

3.4.1.1 External Flood Protection Measures for Seismic Category I Structures. The flooding due to a postulated Main Cooling Reservoir (MCR) embankment breach produces the maximum water level around the power block structures as well as the controlling water elevations for buoyancy calculations. This is also the controlling phenomena in determining the maximum water level at the Essential Cooling Water Intake Structure (ECWIS). Studies and analyses on the MCR embankment have demonstrated that an adequate margin of safety can be maintained for all credible failure mechanisms (Section 2.5.6). Accordingly, mechanistic effects (such as scour and erosion) associated with a postulated failure of the MCR embankment need not be evaluated.

The maximum water level on a vertical face at the south end of the plant structures is El. 50.8 ft mean sea level (MSL), which is El. 22.8 ft above plant grade. This maximum elevation occurs during a quasi-steady-state condition after a breach of the MCR embankment and is based on an instantaneous removal of approximately 2,000 ft of the embankment opposite the power block structures. This maximum elevation occurs on the south face of the Fuel-Handling Building (FHB) of Unit 1. The selection of postulated embankment breach widths and the assumptions made in determining the maximum flood elevations are described in Section 2.4.4.

Total inundation of the Essential Cooling Pond (ECP) occurs only under the condition of MCR embankment breach and does not affect the safe shutdown capability of the plant. The maximum water level calculated to occur at the ECWIS is El. 40.8 ft.

Safety-related structures, systems and components listed in Table 3.2.A-1 are protected against the effects of external flooding by:

1. Being designed to withstand the maximum flood level and associated effects and remain functional (such as seismic Category I structures and the Category I auxiliary feedwater storage tank) or

2. Being housed within seismic Category I structures which are designed as in item 1, above.

Flood protection of safety-related structures, systems, and components is provided for postulated flood levels and conditions described in Section 2.4.

Seismic Category I structures are designed to withstand the maximum flood levels by:

1. Having external walls and slabs of structures designed to resist the hydrostatic and hydrodynamic forces associated with surge-wave runup and steady-state water level.

2. Ensuring the overall stability of the total structure against overturning and sliding due to the hydrostatic and hydrodynamic forces associated with surge-wave runup and steady state water level, and

3. Ensuring that the total structure will not float due to buoyancy forces.

Figure 3.4-1 shows a general section through the plant. Figure 3.4-2 shows the seismic Category I Building maximum steady-state water surface profile, and the corresponding relationship of sill elevations for entrances to seismic Category I buildings.

Table 3.4-1 shows the results of hydraulic loading and buoyancy calculations which were done for the various safety-related facilities. The water depths shown on this table were developed from the maximum water surface elevations presented in Table 2.4.4-3.

An investigation of seismic Category I structures has been made for the flood levels and associated effects as previously described. The design for gross effects upon the structure incorporates safety factors greater than 1.1. All exterior seismic Category I building openings are located above the maximum steady-state flood level or are equipped with watertight doors when located below this profile, except as stated below.

Exceptions to the above-stated design basis for exterior building openings in seismic Category I structures are: (1) the opening for the truck bay in the radwaste loading area of the Mechanical-Electrical Auxiliaries Building (MEAB) and (2) the opening for the rail car access in the spent fuel cask loading area of the FHB. These areas are not protected from flooding because they do not have any safety-related systems and components located near or below the maximum flood level which is required to perform any essential function. In addition, the two areas are separated from the remainder of the building by walls which do not contain openings below the maximum water surface elevation corresponding to their location. The Tendon Gallery Access Shaftcover (TGAS) is provided with a watertight cover to prevent flood waters from entering the MEAB.

The safety-related equipment in the ECWIS is protected from the effects of the design basis flood. The personnel access doors on the west wall are provided with watertight doors; all other doors and openings are above the flood level. The dividing walls and doors between the ECWIS compartments minimize the potential for the propagation of flooding from one compartment to another.

The three maintenance knockout panels in the exterior walls of the Diesel-Generator Building (DGB), which are located below the maximum water surface elevation of 45.0 ft MSL, are watertight and designed for the hydrostatic forces. Each knockout panel allows access to only one of the three separate compartments within the structure, and only one panel may be removed at one time. The dividing walls between the compartments preclude propagation of flooding from one compartment to another.

The maintenance knockout panels in the exterior wall of the room, housing the component cooling water heat exchangers in the MEAB are located below the maximum steady-state water level shown on Figure 3.4-2. These panels are watertight. Since mechanistic effects from the MCR breach need not be evaluated, there is adequate time to replace the knockout panels for the remaining flood events of concern.

All exterior seismic Category I building wall and slab surfaces below grade are waterproofed. This conservatively protects the substructure of seismic Category I buildings from groundwater, which is expected to stabilize between El. 17 ft and 26 ft (1 to 10 ft below grade) after decommissioning of the dewatering system. No waterproofing is provided on exterior wall or slab surfaces above grade to protect against the effects of surge-wave run-up because of its short duration. All construction joints in exterior walls and slabs (except for localized areas of blockouts) are provided with waterstops to the maximum flood level for that location and can withstand hydrostatic and hydrodynamic effects.

All seismic joints between Category I structures contain dual 9-in. water stops capable of withstanding potential seismic and hydrostatic effects. Cracks in concrete are minimized by imposing strict QA and QC procedures on the quality of concrete and construction techniques.

Drains are provided with check valves such that the external flooding would not result in internal flooding through the inadvertent introduction of water through these drains into seismic Category I structures.

The duct banks are sealed so as to prevent backflow into safety-related areas. The cable in the duct banks is designed/specified for submerged installations.

Leakage from groundwater into the FHB is prevented by the use of waterproofing on exterior wall and slab surfaces located below grade. Should groundwater inleakage occur, it is handled by the pumps in the FHB sump, the three-train compartment sumps, and the transfer cart area sump. For Unit 1 only, accumulated groundwater inleakage to the 64 degree tendon buttress area drains through a penetration in the RCB tendon gallery outer wall and is collected in the tendon gallery sump.

Leakage of groundwater into the MEAB is prevented by the use of waterproofing on exterior wall and slab surfaces located below grade. Should groundwater leakage occur, it will be collected in sumps. Discharge from non-radioactive sumps are routed to the reservoir via a circulating water discharge line. Potentially radioactive discharge is pumped to the Liquid Waste Processing System (LWPS).

3.4.2 Analysis Procedures

3.4.2.1 Phenomena Considered in Design Load Calculations. For external flooding, the design basis events considered in design load calculations are as described in Section 3.4.1.

3.4.2.2 Flood-Force Application. The design flood conditions and elevations have been determined from an analysis of the phenomena discussed in Section 3.4.1.1.

In order to establish the controlling load conditions resulting from the embankment breach, both instantaneous surge wave runup as well as the longer term, quasi-steady-state conditions were analyzed. The wave runup condition conservatively assumes that the maximum total force perpendicular to the south face of the plant structures includes a dynamic component in addition to the associated hydrostatic forces. The quasi-steady state condition assumes that only the hydrostatic component contributes to the development of the total force for this case. The latter condition resulted in higher water surface elevations and greater hydraulic loads on power block structures.

The vertical buoyant loading condition is the force equal to the weight of water displaced by a structure. The discussion of lateral and vertical loadings is presented in the following subsections. Table 3.4-1 shows a summary of different lateral loadings at various locations around plant and ECP structures, caused by their respective controlling flood conditions. Procedures used to determine flood loadings are identified in Sections 3.4.2.2.1 and 3.4.2.2.2.

3.4.2.2.1 Lateral Loading:

3.4.2.2.1.1 Lateral Loading on the Power Block Structures – The analysis of the lateral force on the power block structures considered both the instantaneous wave runup and the quasi-steady state conditions. This analysis determined that the maximum total lateral force on the power block structures occurs when the maximum water level is reached during the quasi-steady state condition. Table 3.4-1 shows the controlling lateral forces (hydrostatic) exerted on different power block structures. These lateral forces are treated as triangular loadings on a vertical surface, varying at a rate of 62.4 lb/ft2/ft of structure depth. The procedures used to determine the dynamic and hydrostatic loadings for the above analysis conditions are discussed below:

1. Dynamic Force

The dynamic force on the south side of the power block structures is determined by application of linear momentum principles. The flow from the MCR is assumed to be normal to the south side of the power block structures. Therefore, the dynamic force exerted on the structures can be expressed by the following momentum equation (Ref. 3.4-2):

$$F = p\ Q\ V_o$$

where:

F = dynamic force normal to plant structure
p = density of flow
Q = flow rate
V_o = velocity of flow

The maximum value of pQv_o during surge formation is calculated. This is the contribution of momentum flux to the dynamic force. The contribution of the unsteadiness of momentum field is insignificant.

2. Hydrostatic Force

The lateral hydrostatic force is determined by the following equation (Ref. 3.4-2):

$$F_{Hyd} = \frac{1}{2}\ \gamma_w h^2$$

where:

F_{Hyd} = hydrostatic force, lb/ft of width

h = water depth, ft

γ_w = unit weight of water, lb/ft^3

3.4.2.2.1.2 Lateral Loading on the ECWIS and the South ECP Embankment – The determination of the maximum lateral force on the ECWIS considered both instantaneous and quasi-steady-state conditions. The maximum total force on the ECWIS is a result of the MCR embankment breach discussed in Section 2.4.4.2.2. This force is the result of a water elevation of 41.0 ft mean sea level during the quasi-steady state condition.

Since the south ECP embankment crest elevation is 34.0 ft MSL, it would be overtopped by the flood wave resulting from the MCR embankment breach. The south ECP embankment is designed to withstand the lateral force based on the maximum water elevation resulting from MCR embankment breach.

3.4.2.2.2 Vertical Loading: The roofs of seismic Category I structures are designed to withstand the weight of the accumulated PMP, assuming completely clogged drains (Section 2.4.2.3).

Table 3.4-1 shows the elevations of maximum water surface used for buoyancy calculations. The maximum buoyant force is calculated by assuming that the granular backfill around the structures is completely saturated so that the buoyant force will occur as soon as water arrives at the plant area.

3.4.3 Internal Flood Protection

3.4.3.1 Protection Features. Safety-related systems, components and structures are protected such that the plant can achieve and maintain a safe shutdown condition and prevent unacceptable radiological releases to the environment.

In general, the plant layout arrangement is based on maximizing the physical separation of redundant or diverse safety-related components and systems from each other and from nonsafety-related items. Therefore, there is minimal effect on other systems or components which are required for safe shutdown of the plant or to mitigate the consequence of internal flooding.

Where separation is not feasible, other protection features are employed. These protection features include the following:

- Structural enclosures including watertight doors
- Structural barriers
- Curbs and elevated thresholds
- Seismically designed components

- Hardening
- Orientation
- Equipment leak detection systems
- Floor Drain System

Specifically, watertight doors are designed to withstand the design flood level on either side of the door and to prevent leakage through the door. Structural barriers or spray shields are designed to preclude water spray damage from postulated leakage cracks and sprinkler activation. Curbs and elevated thresholds are designed to prevent leakage from compartments and unsealed cubicles to other areas. Penetration seals through firewalls or radiation barriers of rooms are designed to withstand the flood level on either side of the wall and prevent leakage through the penetration. Class 1E leak detection level instrumentation is provided for the containment spray and safety injection system rooms. (See Section 9.3.3.2.3 for more design information for the leak detection level instrumentation.)

The Floor Drain System is equipped to protect safety-related equipment from the effects of leakage of systems within the building as described in Section 9.3.3. For example, concrete floors are sloped to floor drains located at low points in the same area to facilitate floor drainage and prevent water accumulation. Also safety-related equipment will be protected from unacceptable damage due to flooding caused by reverse flows through the drainage system by either the drain system design or building design features.

3.4.3.2 Internal Flood Analysis. Methodology used in analysis of the effects of high energy line breaks is discussed in Section 3.6. Flooding effects analyses are contained in Appendix 3.6.B for postulated high energy line breaks. For example, the containment flooding analysis has shown that the maximum volume of water discharge to the RCB occurs as a result of a Loss of Coolant Accident (LOCA), and water from the RCS, accumulators and the RWST is assumed to spill onto the RCB floor.

Reviews of internal flooding from other sources (tank ruptures, moderate energy cracks, etc.) within the following buildings are performed to assure the essential functions of affected safety-related systems, components and structures necessary to achieve and maintain a safe shutdown condition or that the appropriate combinations of the above protective measures are used:

- Mechanical and Electrical Auxiliary Building including the Isolation Valve Cubicle
- Diesel Generator Building
- Fuel Handling Building
- ECW Intake Structure

The following is an example of the analysis methods of Section 3.4.4 used to determine the appropriate protection method within the Isolation Valve Cubicle. Similar spray and flooding evaluations for all safety-related buildings have been performed. A description of the flooding evaluation results is provided in Section 3.4.4 and Appendix 3.6.B.

Appendix 3.6.A, Section 3.6.A.1 indicates that watertight doors were used to maintain the complete separation between trains of the IVC to preclude adverse flooding effects from postulated high energy line breaks.

Due to the lift-off or vent panel design on the IVC roof (designed to relieve the pressure buildup following postulated pipe breaks described in Appendix 3.6.A) internal flooding due to rainfall associated with tornados has been evaluated.

Analyses, in accordance with the methods of Section 3.4.4, have been performed to determine the flood level in the pump cubicles. Since the lift-off panels on all four cubicle compartments are affected by the assumed tornado depressurization loading, all trains of the auxiliary feedwater pump cubicles are impacted. Based on the results of the flood level analyses, additional curbs are provided to channel rainwater away from the auxiliary feedwater pump cubicles in order to preclude unacceptable consequences. The floor drains directly above these pump cubicles will be permanently capped to prevent rainwater from entering these areas.

In the case of the Spent Fuel Pool Cooling System, a postulated moderate energy line crack in a section of Component Cooling Water or Spent Fuel Pool Cooling System piping which is common to both trains of the Spent Fuel Pool Cooling System will result in the inability to maintain Spent Fuel Pool Cooling via the Spent Fuel Pool Cooling System for a period of time. In this event, the temperature of water in the pool will increase until boiling occurs. Fuel pool boiling may also occur if a pipe crack disables one train of the Spent Fuel Pool Cooling system while a single failure results in the loss of the redundant train. These events have been analyzed and the results were found to be acceptable (Section 9.1.3.3.4).

3.4.4 Internal Flood Analysis Procedures

3.4.4.1 Sources of Internal Flooding. The internal flooding analysis assures that safety-related systems, structures and components are not prevented from performing their essential functions following the postulated failure. The sources of flooding are:

- Moderate energy lines with through wall cracks
- Tank ruptures
- High energy line breaks
- Activation to the fire protection system

3.4.4.2 Considerations and Assumptions. The flooding analysis assesses the maximum flow of fluid from the postulated break, crack, or sprinkler flow that possesses the maximum fluid discharge in a specific area. The maximum time for flood will vary according to the particular case being analyzed. Operator actions in the main control room to mitigate the consequences are assumed to be initiated a minimum of 10 minutes after control room indication is available to show that action is required. Operator response time for actions outside the control room is assumed to be 30 minutes after control room indication. An operator response time shorter than identified above is used if shown feasible by analysis.

If a postulated break of a pipe causes other piping to fail as a consequence of the initial break, then the flow from the secondary piping is considered in the flooding analysis.

Flooding caused by activation of the Fire Protection System is not considered concurrently with other design basis accidents or events.

A single, limiting independent active component failure is considered in conjunction with the effects of flooding. Single failures are not assumed in a system or component which normally operates at the time of failure initiation and which also functions to mitigate the break event, provided the system or component is designed to seismic Category I requirements and is qualified for the environment associated with the break.

If the postulated failure results in automatic separation of the turbine generator from the power grid, then offsite power is assumed unavailable unless the assumption of loss of offsite power is not conservative (e.g., termination of flooding due to loss of power to a pump). Power restoration is assumed after 24 hours.

For calculating outflow in postulated line failures, the normal operating pressure and temperature are utilized as the initial thermodynamic conditions. The volume occupied by equipment in a room is considered when performing the flooding analysis for the water height. The occupied volume of this equipment is subtracted from the total volume of the room. Appropriate credit for gravity drains and the volume occupied by sumps is considered in flood height determinations.

Postulated post-SSE failures of nonseismic Category I fluid systems are considered individually in the flooding analyses.

Postulated flooding caused by failures of nonseismic Category I non-tornado protected tanks in the yard and inside seismic Category I buildings shall not result in failure of a safety-related system to perform its essential function.

Each tank rupture is evaluated as follows:

1. Instantaneous release of tank fluid capacity for nonseismic Category I tanks.

2. Fluid flow through an area of a through wall crack equal to one-half the thickness by one half the outside diameter of the largest fluid discharge connection to seismic Category I tanks.

The use of nonseismic Category I system in mitigating the consequence of postulated piping failure (other than a main steam system piping failure) outside the containment is clarified in the following paragraphs:

1. For nonseismic Category I piping failures, it is assumed that a safe shutdown earthquake could be the cause of the failure. Therefore, only seismic Category I equipment can be used to mitigate the consequences of the failure and bring the plant to a safe shutdown.

2. A postulated failure in seismically qualified portions of piping systems is not assumed to be seismically induced. Propagation of the failure to failures of nonseismically qualified equipment is not assumed. Nonseismic Category I equipment can be used to bring the plant to a safe shutdown following a postulated failure in seismically qualified piping, subject to power being available to operate such equipment and providing the equipment is qualified for the environment resulting from the piping failure.

In accordance with the above criteria, credit is taken for the use of redundant nonseismic, nonsafety grade sump level instruments for the detection of flooding not caused by a seismic event. No credit is taken for the use of such instruments following a seismic event until their functionability has been verified. No credit has been taken for the use of nonseismic sump pumps following a seismic event.

For areas where redundant water level indications are not available for the detection of flooding resulting from a piping failure, regular inspections during normal operation by operations personnel is being implemented. The walkdowns for flood detection are being implemented for the following areas.

- Common area north of the auxiliary feedwater pump rooms. This area has one sump which is equipped with a single level switch and a local level alarm. Therefore, flooding in this area will not be alarmed in the control room. Moderate energy lines which are potential flood sources for this area are:

    1. Auxiliary feedwater pump suction lines

    2. Sump pump discharge lines

    3. Fire protection lines

    A crack on an auxiliary feedwater pump suction line will be detected by a level decrease in the auxiliary feedwater storage tank which does not have automatic makeup and which has three redundant safety-related level transmitters with indication in the control room. Failure of a sump pump discharge line will not result in significant flooding because of the limited amount of water which will drain into the sumps during normal operation. A crack on a fire protection line will be detected if the crack flow is over 30 gal/min because the resultant drop in the fire protection header pressure will cause a main fire pump to automatically start, thus alerting the control room operators of the loss of water from fire protection system piping. A crack flow of less than 30 gal/min may not be detected because the fire jockey pump will maintain system pressure. However, because of the low crack flow rate, the flood level in this area will not reach the worst case flood level unless the crack flow is allowed to continue for more than 36 hours (based on a crack flow of 30 gal/min, smaller cracks will take longer to reach the same level). Because this area will be inspected regularly during normal operation, such small cracks on the fire protection system piping will be detected long before flooding reaches an unacceptable level.

CN-3171

- Tendon Access Gallery

  The tendon access gallery has a sump equipped with a nonsafety-related level switch with a local alarm. Therefore, flooding in this area will not be alarmed in the control room. The sources of flooding in this area are limited to a 2-inch potable water line used for sump pump testing and the EAB floor drain lines which empty into this area. For Unit 1 only, another source is accumulated groundwater inleakage to the 64 degree tendon buttress area which drains through a penetration in the RCB tendon gallery outer wall. There is no safety-related equipment located in the tendon access gallery.

  The free volume of the tendon access gallery is over 30,000 ft^3. A crack on the 2-inch potable water line will not discharge enough water to completely fill this area unless it is allowed to continue for over 7 days. Significant discharge of water from the EAB floor drain lines is possible in the case of a moderate energy line crack or a fire in the EAB. However, such discharge will be terminated when the flood source in the EAB is isolated. Fire protection system inadvertent actuation or cracks in the fire water lines will be detected by the start of the main fire pumps. Small flood sources in the EAB which may not be detected and isolated quickly will correspondingly have a low flow rate. The groundwater leakage which drains from the Unit 1 tendon buttress area also has a relatively small flow rate. Given the large volume of the tendon access gallery, such small flood sources will not completely fill this area before they are detected by the walkdown which will be performed at an interval of no more than 28 hours. So that operations personnel will not be required to go to the lower elevation of the tendon gallery, a sounding device may be used to physically detect the presence of water. Because the floor of the tendon access gallery is at El. (-)36 ft-9 in. which is 46 ft-9 in. below the EAB floor slab, flooding in the tendon access gallery will not impact the EAB.

In addition, water level instruments in the following areas of the Mechanical Auxiliary Building (MAB) are not seismically qualified. Therefore, walkdowns for flood detection are being implemented following a seismic event. The maximum interval between these walkdowns is 2 hours. The walkdowns would continue until adequate sump level instruments and alarms are shown to be functional.

- Refueling water storage tank compartment or reactor makeup water storage tank compartment

- Recycle holdup tanks 1A and 1B compartments

- Containment penetration area at El. 10 ft-0 in. or any one of the areas containing MAB sumps 1 through 4

Based upon the above inspection intervals, maximum flood levels were determined for the affected areas of the plant and all submerged safety-related components were identified. This evaluation has determined that no essential components required for safe shutdown will be impacted.

REFERENCES

Section 3.4:

3.4-1 Not used.

3.4-2 Streeter, Victor L., “Fluid Mechanics”, 3rd Edition, McGraw-Hill Book Company, Inc., 1962.

TABLE 3.4-1

FLOOD LOADS FOR CATEGORY I STRUCTURES

	REACTOR CONTAINMENT BUILDING				MECH. & ELECT. AUXILIARIES BUILDING				FUEL HANDLING BUILDING				DIESEL GENERATOR BUILDING				ESSENTIAL COOLING WATER INTAKE STRUCTURE				AUXILIARY FEEDWATER CONDENSATE STORAGE TANK			
DIRECTIONS	N	S	E	W	N	S	E	W	N	S	E	W	N	S	E	W	N	S	E	W	N	S	E	W
EL. A				28.0	28.0	28.0	28.0	28.0		28.0	28.0	28.0	28.0	28.0	28.0	28.0	25.0	25.0	10.0	34.0	28.0	28.0	28.0	28.0
H_1				21.0	16.5	23.0	21.0	23.0		23.0	23.0	23.0	17.0	17.0	17.0	17.0	16.0	16.0	31.0	7.0	22.0	22.0	22.0	22.0
P_S				1.31	1.03	1.44	1.31	1.44		1.44	1.44	1.44	1.06	1.06	1.06	1.06	10.0	10.0	19.3	0.44	1.37	1.37	1.37	1.37
F_T				13.8	8.5	16.5	13.8	16.5		16.5	16.5	16.5	9.0	9.0	9.0	9.0	8.0	8.0	30.0	1.5	15.1	15.1	15.1	15.1
H_2				7.0	5.5	7.7	7.0	7.7		7.7	7.7	7.7	5.7	5.7	5.7	5.7	5.3	5.3	10.3	2.3	7.3	7.3	7.3	7.3
BUOY EL				49.0	44.5	51.0	49.0	51.0		51.0	51.0	51.0	45.0	45.0	45.0	45.0	41.0	41.0	41.0	41.0	50.0	50.0	50.0	50.0

Legend (Refer to Figure 3.4-3):

EL. A = Ground Elevation (feet above MSL)

P_S = Lateral hydrostatic pressure in KSF

H_1 = Height of maximum water level attained at face of structure in feet

H_2 = Point of application of resultant force, F_T in feet

F_T = Resultant force in K/FT

3.5 MISSILE PROTECTION

This section describes the missile protection design bases for seismic Category I structures, systems, and components. Seismic Category I structures, systems, and components and their safety classifications are identified in Section 3.2. Missiles considered are those which could result from: a plant-related failure/incident, including failures within and outside of the Reactor Containment Building (RCB), environmentally generated missiles, and site proximity missiles. Included in this section are descriptions of the structures, shields, and barriers which are designed to withstand missile effects, the possible missile loadings, and the procedures by which each barrier is designed to resist missile impact.

To reduce the probability of unacceptable consequences related to missile impact, key backup and/or redundant components and systems have been physically separated and shielded so that a single missile is incapable of negating the redundant functions. In addition, essentially all seismic Category I components are housed in seismic Category I structures or analysis is performed to demonstrate that external missiles have an acceptably low probability of striking them.

The following criteria were adopted for assessing the plant's capability to withstand the missiles postulated in Sections 3.5.1.1 and 3.5.1.2:

1. No perforation of the RCB (i.e., no loss of leaktightness).

2. Assurance that the plant can achieve and maintain a safe shutdown condition.

3. Offsite exposure within 10CFR100 guidelines for missile damage which could result in activity release.

3.5.1 Missile Selection and Description

Wherever possible, component and system design precludes the generation of missiles. This is achieved by suitable choice of materials, use of normal and faulted stress levels, and system and component characteristics which avoid missile-producing effects even under faulted conditions. For example, valve stem missiles from manual gate or globe valves are precluded by using valves with backseats in high pressure systems.

Wherever possible, systems and components identified as potential missile sources are arranged so the postulated missile would impact on a structure or component capable of withstanding the impact.

Barriers are provided for missiles which cannot be oriented to take advantage of other structures and which could cause failure of essential safety-related structures or components. These barriers are designed to contain or deflect the missiles from the essential safety-related component without generating any secondary missiles.

Wherever possible, equipment is located so as to take advantage of walls and other structures (provided to meet other functional requirements) to separate essential components from potential missile sources.

Table 3.5-2 lists and describes the barriers utilized for missile protection.

3.5.1.1 Internally Generated Missiles Outside the Containment. Seismic Category I structures, systems, and components outside the Containment whose failure could result in radiological consequences in excess of 10CFR100 guidelines or which are required for attaining and maintaining a safe shutdown during normal or accident conditions are listed in Table 3.5-1. External missile protection provisions and references to applicable system descriptions and drawings that demonstrate separation and independence are listed in Table 3.5-1. Internal missile protection categories for safety-related systems, structures and components are indicated in Table 3.5-1. Protection requirements from internal missile sources are described below for the following potential missile sources:

- High-pressure systems
- Rotating machinery
- Gravitational missiles
- Compressed air/gas cylinders

Systems outside the Containment were reviewed to determine sources of missiles. Compressed air/gas cylinders are either separated from safety-related components in cubicles or subcompartments within the structure, not located within the structures which house safety-related systems or they are restrained. The results of this review are discussed in the following section.

3.5.1.1.1 High-Pressure Systems: Valve bonnets and stems, thermowells and tanks are the potential missiles associated with high-pressure systems outside the Containment.

Temperature detectors installed in high energy piping are evaluated as potential missiles where they are only attached by a threaded connection. Where they are attached by a threaded connection with a seal weld, the seal weld prevents the connection from disengaging because of vibration, cyclical stresses, etc., and these detectors are not postulated as missiles. Where they are attached by welding, the design strength of the completed weld is at least equal to or greater than the base materials and, therefore, these detectors are not postulated as missiles. In addition, because of the spatial separation of redundant safety-related equipment, a small missile such as a detector, assuming the circumferential weld fails completely, is not likely to hit redundant safety-related equipment.

Two types of valve components, valve stems and valve bonnets, are potential missiles. Valves in high-pressure systems have been reviewed as potential missile sources. The provisions that valves have bolted bonnets or secondary retention devices, and that they be designed to ASME III requirements effectively eliminates credible sources of valve component missiles.

Valves of American National Standards Institute (ANSI) 900 psig rating and above, constructed in accordance with Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, are pressure seal bonnet type valves. For pressure seal bonnet valves, valve bonnets are prevented from becoming missiles by the retaining ring, which would have to fail in shear, and by the yoke, which would capture the bonnet or reduce bonnet energy.

Because of the highly conservative design of the retaining ring of these valves (safety factors in excess of 8 may be used), bonnet ejection is highly improbable and hence bonnets are not considered credible missiles.

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

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

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

Valves with threaded bonnet studs are not utilized in high energy piping and thus are of no concern as potential missiles.

Valves in high pressure systems have been reviewed. As a result of this review it has been determined that no failure associated with a single valve part can result in the generation of a missile.

Pressurized tanks in high pressure systems are either not located within the structures which house safety-related systems or they are separated from safety-related components in cubicles or subcompartments within the structure.

3.5.1.1.2 Rotating Machinery: Potential missile sources associated with rotating machinery were identified as:

- Motor-driven pumps and compressors
- Turbine-driven pumps
- Heating, ventilating, and air conditioning (HVAC) fans
- Diesel generator (DG) turbocharger rotors
- Motor generator set flywheels

Rotating equipment evaluated as potential sources of missiles were either determined incredible (based on supplier certifications) as missile sources, evaluated using the formulas given Section 3.5.3, or the effect of loss of the potential missile targets on the ability to shutdown safely was reviewed. A summary of rotating equipment considered as potential missile sources is given in Table 3.5-16.

Missile selection was based on the following considerations:

1. Rotating equipment that is operated during normal plant conditions is capable of generating a missile.

2. The energy of a rotating part in a high energy system associated with 120 percent overspeed is assumed for component failure unless analysis is performed to indicate otherwise. For moderate energy systems, evaluations are based on missiles postulated to occur at normal speeds.

3. Determination of whether the energy of the missile is sufficient to perforate the protective housing. For example, electrical motors are not considered potential missile sources due to their cast iron housing. The housing itself is capable of withstanding internal faults such as cooling fan break down or armature disintegration. The following are not potential missile sources:

    a. There are four turbine-driven pumps, of two types: the turbine-driven auxiliary feedwater pump and the three turbine-driven steam generator (SG) main feed pumps. The main feed pumps and their drive turbines are protected from overspeed by redundant overspeed trips. A single overspeed trip is provided on the auxiliary feedwater pump drive turbine. These pumps and turbines are not considered to be a source of missiles.

    b. The diesel generators (DGs) are designed to withstand overspeeds of 125 percent; redundant mechanical and electrical overspeed trips operate at 110 percent overspeed. The only portion of the diesels considered to be a credible source for postulated missiles is the turbocharger, which is not speed controlled and operates at high rpm. The turbocharger rotors weigh 270 pounds and are mounted on the diesels. In the event of failure, only one DG unit would be affected since each is separated from adjacent units by 2-foot-thick reinforced concrete walls which would contain any turbocharger missile.

    c. Motor generator (MG) set flywheels were reviewed to determine missile generation potential. The fabrication specifications of the MG set flywheels control the material to meet American Society for Testing and Materials (ASTM) A533-70a, Grade B, Class I, with inspections in accordance with MIL-I-45208A and flame-cutting and machining operations governed to prevent flaws in the material. Nondestructive testing for nil-ductility (ASTM-E-208), Charpy V-notch (ASTM A593-69), ultrasonic (ASTM A578-71b and A577-70a), and magnetic particles (ASME Section III, NB2545) has been performed on each flywheel material lot. In addition to these requirements, stress calculations have been performed consistent with guidelines of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel

(B&PV) Code, Section III, Appendix A to show the combined primary stresses due to centrifugal forces and to show that the shaft interference fit does not exceed one-third of the yield strength at normal operating speeds (1,800 rpm) and does not exceed two-thirds of the yield strength at 25 percent overspeed. However, no overspeed is expected for the following reason: The flywheel weighs approximately 1,300 lbs and is 35.26 inches in diameter by 4.76 in. wide. The flywheel mounted on the generator shaft, which is directly coupled to the motor shaft, is driven by a 200-hp, 1,800-rpm synchronous motor. The torque developed by the motor is insufficient for overspeed. Therefore, there are no credible missiles from the MG sets.

4. Some equipment configurations provide an unprotected aspect of the rotating component most likely to eject a missile, i.e., they may provide an opportunity for a missile to be released without impacting the component housing. For example, centrifugal fans may have an open discharge scroll without a ducted exhaust. The connected ductwork on other centrifugal fans may not be as thick as the evaluation shows is needed to prevent perforation by an oblique or perpendicular missile impact.

5. Single failure considerations are similar to those used in the pipe rupture analysis (Section 3.6.1.1).

3.5.1.1.3 Gravitational Missiles: Virtually the only significant gravitational missiles would be from overhead cranes. As discussed in Section 9.1.4, overhead cranes either have interlocks or are single-failure-proof or are administratively controlled so that missiles resulting from dropped loads are not considered further. In addition, missiles could result from a crane derailing and falling. However, overhead cranes were designed with clamping devices to prevent derailing. Jib cranes are bolted to their seismic platform which is mounted to the top of the secondary shield walls. Therefore, no generation of missiles is expected from derailment or falling from an overhead crane. Appropriate measures (such as interlocks, special slings, etc.) have been provided to prevent accidental drop of a heavy load that could impact nuclear fuel, safety-related equipment, or components whose failure could result in radiological consequences exceeding 10CFR100.

3.5.1.1.4 Compressed Air/Gas Cylinders: Compressed air or compressed gas cylinders not part of a connected system have been evaluated for their potential to damage essential safety-related equipment. Most pressurized cylinders are located in areas outside structures housing safety-related equipment. Some are located in cubicles or compartments which separate them from safety-related equipment. Two types of high pressure compressed gas cylinders may be located in areas which contain essential safety-related equipment: portable carbon dioxide fire extinguishers and miscellaneous gas cylinders used to support chemical analyses. These cylinders will be secured as appropriate, in vehicle-type brackets or seismically designed racks and oriented so that an ejected fitting could not strike an essential safety-related component.

3.5.1.2 Internally Generated Missiles Inside the Containment. Systems and components inside the Containment whose failure could result in radiological consequences in excess of 10CFR100 guidelines or which are required for attaining and maintaining a safe shutdown during normal or accident conditions are listed in Table 3.5-1. No missile protection provision is necessary for the postulated missiles described in the following sections.

Potential sources of missiles are:

- High-pressure systems
- Rotating machinery
- Gravitational missiles
- Secondary missiles
- Compressed air/gas cylinders

3.5.1.2.1 High-Pressure Systems – Catastrophic failure of the reactor vessel, SGs, pressurizer, reactor coolant pump (RCP) casings, Safety Injection (SI) accumulators, and piping leading to generation of missiles is not considered credible. Massive and rapid failure of these components is incredible because of the material characteristics, inspections, quality control during fabrication, erection and operation, conservative design, and prudent operation as applied to the particular component.

Components that nevertheless are considered to have a potential for missile generation inside the Containment are:

1. Control rod dive mechanism (CRDM) housing plug, drive shaft, and drive shaft and drive mechanism latched together
2. Valves
3. Temperature and pressure sensor assemblies
4. Pressurizer heaters

These potential missile sources are discussed in the following sections.

3.5.1.2.1.1 Control Rod Drive Mechanism Missiles – Gross failure of a CRDM housing sufficient to allow a control rod to be rapidly ejected from the core was not considered credible for the following reasons:

1. The Unit 1 and Unit 2 replacement CRDM housings were hydrostatically tested during fabrication in conjunction with ASME Section III hydrostatic testing of the replacement head.
2. The CRDM housings are made of type 316 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

For the Unit 1 and Unit 2 replacement CRDMs, the rod travel housing and the top cap on the replacement CRDMs are now one integral piece. The separate cap has been eliminated and cannot act as a missile. However, for the original CRDMs, it was postulated that the top plug on the CRDM will become loose and it will be forced upward by the water jet. The following sequence of events was assumed: The drive shaft and rod control cluster (RCC) are forced out of the core by a differential pressure of 2,500 psi across the drive shaft. The drive shaft and RCC, latched together, are assumed fully inserted when the accident starts. After approximately 14 ft of travel, the RCC spider hits the underside of the upper support plate. Upon impact, the flexure arms in the coupling joining the drive shaft and RCC fracture, completely freeing the drive shaft from the RCC. The RCC would be completely stopped by the upper support plate; however, the drive shaft would continue to be accelerated upward to hit the missile shield.

The CRDM missiles are summarized in Table 3.5-3. The velocity of the missiles was calculated by balancing the forces due to the water jet. No spreading of the water jet was assumed.

3.5.1.2.1.2 Valves – Valve bonnets and stems have been eliminated on the same basis as valve missiles outside Containment. Refer to Section 3.5.1.1.1.

3.5.1.2.1.3 Temperature and Pressure Sensor Assemblies – Temperature and pressure sensor assembly (inside Containment) missiles are treated in the same manner as those outside Containment. Refer to Section 3.5.1.1.1.

3.5.1.2.1.4 Pressurizer Heaters – It was assumed that the pressurizer heaters could become loose and become jet-propelled missiles. The missile characteristics of the pressurizer heaters are given in Table 3.5-4. A 10-degree-expansion, half-angle water jet was assumed.

3.5.1.2.2 Rotating Machinery – The RCP flywheel was not considered a source of missiles for the reasons discussed in Section 5.4.1.

Missile selection is based on the considerations discussed in Section 3.5.1.1.2. A summary of rotating equipment considered as potential missile sources is given in Table 3.5-17.

3.5.1.2.3 Gravitational Missiles – The consequences of a load drop have been studied. The drop of the most critical load lifted by the polar crane does not have unacceptable consequences. Clamping devices prevent the crane from derailing and generating missiles. Appropriate preventive measures (such as interlocks, special slings, etc.) have been identified to prevent accidental drop of a heavy load that could impact nuclear fuel, safety-related consequences exceeding 10CFR100.

3.5.1.2.4 Secondary Missiles – Orientation of the possible missile sources and the design of the barriers is such that there is no possibility of generation of secondary missiles. (Refer to Section 3.5.3 for additional information regarding critical wall thickness to prevent spalling.)

3.5.1.3 Turbine Missiles.

Turbine missiles have been evaluated not to be a credible threat for the STP design basis.

3.5.1.4 Missiles Generated by Natural Phenomena. Flooding, hurricanes, and tornadoes are the only three types of natural phenomena which could generate missiles at STPEGS.

Water-borne missiles would have a maximum velocity of 20 ft/sec, corresponding to the conservatively calculated maximum water speed resulting from a postulated failure of the Main Cooling Reservoir (MCR) embankment. Such missiles could consist of waterborne debris such as automobiles, utility poles, wooden planks, etc. The effects from such missiles are considerably less severe than the effects of the postulated tornado missiles.

Missiles resulting from hurricane winds could be postulated to be similar to the types of missiles generated by tornadoes; however, due to the lower hurricane wind speeds, the effects would be less severe than the effects of tornado-generated missiles. Tornado-generated missiles are used as design basis missiles for STPEGS. A maximum tornado wind speed of 360 mph consistent with a Region I design basis tornado of RG 1.76 (April 1974) is used to calculate the missile velocities. The design parameters for tornado missiles are summarized in Table 3.5-9.

Structures, systems, and components whose failure could prevent safe shutdown of the reactor or result in significant uncontrolled release of radioactivity are protected from such failure due to design tornado wind and missile loading by the following methods:

1. Structure or component is designed to withstand tornado wind loading or tornado missile.

2. Component is housed within a structure which is designed to withstand the tornado wind loading and tornado missile loading.

The only exceptions to the above are the Isolation Valve Cubicle (IVC) roof, MEAB HVAC dampers, the AFW Pump Recirculation Piping and the Diesel Generator exhaust, where the probability of a tornado missile strike is demonstrated by analysis to be much less than 1 x 10^{-7} per year. CN-3159

The only safety-related components located outdoors are the Auxiliary Feedwater Storage Tank (AFST) and portions of the Essential Cooling Water System (ECWS). See Section 3.8.4 for a description of the AFST and Section 9.2.1.2 for a description of the ECWS. The design of STPEGS is such that the structures, systems, and components specified in the appendix to the guide are protected against tornadoes and tornado missiles. As a result, STPEGS is in compliance with RG 1.117. Information on barriers used to protect the principal systems is given in Table 3.5-10.

3.5.1.5 Missiles Generated by Events Near the Site. As discussed in Section 2.2.3, missiles originating from events near the site, such as from explosions, do not impact safety-related structures or components and do not constitute design basis events.

3.5.1.6 Aircraft Hazards. As discussed in Section 2.2.2, there are no airports within 10 miles with greater than 500 d^2 operations per year or farther than 10 miles with greater than 1,000 d^2 operations per year (d is the distance to the airport); therefore, aircraft activities from nearby airports do not constitute a hazard to STPEGS.

The only nearby military aviation activity was flight route OB-19, which was used by the U.S. Air Force and Navy for low-level navigation-bombing training flights for jet aircraft, but the route was cancelled as of January 30, 1975. In fact, the route was not used for three years prior to that. Thus, there is no hazard to STPEGS from military aviation activity.

There is one low-level federal airway within 2 miles of the plant. The centerline of V70 has a closest approach of approximately 5 miles. V20S, which previously coincided with V70 in this area, has been discontinued.

A hazard analysis was performed using the following approach:

$$P_{FA} = C \times N \times A/W$$

where:

P_{FA} = probability per year of an aircraft crashing into the plant from the airway

C = in-flight crash rate per mile for aircraft using the airway

N = number of flights per year along the airway

A = effective area of plant in square miles

W = width of airway plus twice the distance from the airway edge to the site when the site is outside the airway (in miles)

Federal Aviation Administration data for 1976 show that, during the peak 24-hour period, there were 16 flights below 17,000 ft in altitude and 18 flights above 17,000 ft in altitude on V70 in the vicinity of STPEGS. It has been suggested that the flights below 17,000 ft can be characterized as general aviation and those above 17,000 ft can be characterized as U.S. air carrier (Ref. 3.5-10). A 1983 survey of flights in the area indicates there are approximately 25 flights per day within 5 miles of the site (Ref. 3.5-30).

Bases on this, the above approach has been modified as follows:

$$P_{FA} = C_1N_1A/W + C_2N_2A/W$$

where the subscript 1 denotes general aviation and the subscript 2 denotes U.S. air carrier.

The suggested value of

$$C_2 = 3 \times 10^{-9} \text{ accidents/mile}$$

for the U.S. air carrier in-flight accident rate was used.

A conservative estimate of the general aviation in-flight crash rate per mile was obtained by using the ratio of relevant accidents to total miles flown for the period of 1972 to 1976. This information is presented in Table 3.5-11. General aviation statistics were reviewed for 1977 to 1981, (Ref. 3.5-31), but the information no longer presents data in accidents per mile. In addition, general aviation aircraft accident data no longer includes air taxi accidents. Table 3.5-12 shows a decrease in total accidents from 1972 to 1981, while the total hours flown increased nearly 50 percent. After reviewing the later data, it was decided to utilize the 1972 to 1976 data since it remains conservative. Therefore from Table 3.5-11,

$C_1 = 1.8 \times 10^{-7}$ general aviation accidents/mile

The number of flights was determined to be

$N_1 = 4{,}563$ flights/year

$N_2 = 4{,}562$ flights/year

Twice the distance from the plant to the airway center line is

$W = 10$ miles

The effective area of the plant is taken to be the plan area plus the shadow area plus a slide area. The plan area is approximately .0095 mi^2. The shadow area is conservatively calculated to be .013 mi^2 by assuming:

- A conservatively shallow descent angle of 10 degrees
- Impacting aircraft approach from the east (the most conservative direction)
- No overlap of shadow areas and plan areas between units

The slide area is conservatively assumed to be 50 percent of the combined plan and shadow areas. This leads to a total effective area of

$A = .034\ mi^2$

Then it is computed that

$P_{FA} = 2.8 \times 10^{-6}$ per year from general aviation +

4.6×10^{-8} per year from U.S. air carriers

The overall probability of an aircraft crash from U.S. air carrier traffic on V70 does not pose a significant hazard to STPEGS.

The probability of an aircraft crash occurring from general aviation traffic on V70 is about one order of magnitude greater than the acceptance criteria of 10^{-7} per year for radiological consequences greater than the guidelines of 10CFR100. However, general aviation aircraft are light (usually less than 12,500 pounds) and would pose a hazard to plant safety only in the event of striking vulnerable plant areas (e.g., a door or an equipment hatch to a safety-related structure). Since such vulnerable areas constitute a small fraction of the effective area, a reduction of approximately more than an order of magnitude would result.

It is concluded that there is no significant hazard to STPEGS from aircraft traffic on V70.

3.5.2 Systems To Be Protected

Systems to be protected from internal missiles and the protection measures used are identified in Sections 3.5.1.1 and 3.5.1.2.

The only externally generated missiles for which protection is required are tornado missiles, as discussed in Section 3.5.1.4.

Barriers used missile protection are listed in Table 3.5-2.

3.5.3 Barrier Design Procedures

Barriers are designed to withstand the effects of missile impact. The barriers are designed or checked to assure that a missile strike does not cause scabbing. The overall effects are evaluated by the response of the structure or target and portions thereof to missile impact. Missiles are assumed to strike the barriers normal to the surface, and the axis of each missile is assumed to be parallel to the line of flight. These assumptions result in a conservative estimate of missile effect to barriers.

3.5.3.1 Local Damage Prediction. Predication of local damage, i.e., damage in the immediate vicinity of the impact area, includes estimating the depth of penetration, minimum thickness required to prevent perforation, and minimum thickness required to preclude spalling.

3.5.3.1.1 Reinforced Concrete Barriers: The depth of penetration of a missile (excluding turbine-generated missiles) into a reinforced concrete barrier is calculated by the modified Petry formula, as set forth in Reference 3.5-13. Depending upon the slab thickness penetration depth ratio, α', the following expressions are used.

$$D = \begin{cases} 12KA_p \log_{10} \quad 1+\dfrac{V^2}{215{,}000} & for\ \alpha' \geq 3 \\ 12KA_p \log_{10} \quad 1+\dfrac{V^2}{215{,}000} \quad 1+e^{[-4(\alpha'-2)]} & for\ \ 2 \leq \alpha' < 3 \end{cases}$$

where:

D = penetration depth, in.

$$K = 0.00476 \frac{3{,}200}{(1+11p)f'_c} = \text{material parameter}$$

A_p = W/A_c = sectional pressure, lb/ft^2

V	=	impact velocity, ft/sec
W	=	missile weight, lb
A_c	=	missile contact area, ft^2
α'	=	t/D = wall thickness/penetration depth
t	=	wall thickness, ft
p	=	total percent of reinforcement on all faces in all directions
f'_c	=	compressive strength of concrete, psi
e	=	base of Napierian logarithms

When the Petry formula was used for tornado missile analysis, minimum thickness of the concrete barrier has been designed as twice the penetration depth in order to prevent perforation and spalling of the barrier.

In some instances, the following relationships (Ref. 3.5-28) were used to estimate the concrete element thickness for threshold of spalling:

For solid steel missiles:

$$T_s = 15.5 \frac{W^{0.4} \; V_s^{0.5}}{\sqrt{f'_c} \; D^{0.2}}$$

For steel pipe missiles:

$$T_s = \frac{5.42 W^{0.4} \; V_s^{0.65}}{\sqrt{f'_c} \; D^{0.2}} \qquad 12 \leq \frac{r}{t}$$

where:

T_s	=	thickness for threshold of spalling, in.
W	=	missile weight, lbs
D	=	missile diameter, in.
f'_c	=	concrete strength, psi
V_s	=	missile striking velocity, ft/sec
r	=	pipe outside radius, in.

t = pipe wall thickness, in.

The minimum thickness criteria of SRP Section 3.5.3 were satisfied. The minimum thickness of the concrete barrier (both walls and roofs) provided for all Category I structures to resist the effects of postulated tornado winds and missiles is 2 ft, except for the Auxiliary Airlock Shield structure roof which is 1 ft. The integrity of this 1 ft structure has been analyzed and determined to provide the necessary protection for missile impact.

3.5.3.1.2 Steel Barriers: Steel barriers are designed to preclude perforation by missiles. A modified form of the BRL formula (Ref. 3.5-29), shown below was used to determine the threshold thickness of perforation.

$$T_p = \frac{(E_k)^{2/3}}{672D}$$

where:

$$E_k = \frac{M_m V_s^2}{2}$$

and:

T_p = steel plate thickness for threshold of perforation, in.

E_k = missile kinetic energy, ft-lbs

M_m = mass of the missile, lb-sec^2/ft

V_s = missile striking velocity, ft/sec

D = missile diameter, in.*

* In considering the tornado missile spectrum, a wood plank was not taken into consideration in calculating barrier thickness with the above formula, which is valid for non-deformable missiles. It is not considered credible to assume that the wood plank could penetrate without disintegrating a steel barrier of sufficient thickness to resist the rest of the missile spectrum.

The design thickness to prevent performation, t_p, was taken, as a minimum 15 percent greater than the predicted threshold value:

$$t_p \geq 1.15T_p$$

where:

t_p = design thickness to preclude perforation, in.

3.5.3.1.3 Composite Barriers: Analysis of missile barriers composed of several elements involves the determination of the residual velocity after perforation of one element. This value is then used as the striking velocity on the next element. The minimum thickness requirement for concrete and steel (see Sections 3.5.3.1.1 and 3.5.3.1.2) govern the design of the innermost element.

The following equation (Ref. 3.5-6) is used to calculate the residual velocity of the missile after perforation of an element:

$$V_r = \begin{cases} (V_s^2 - V_p^2)^{1/2} \text{ for } V_p \le V_s \\ 0.0 \quad \text{for } \; V_p \ge V_s \end{cases}$$

where:

V_r = residual velocity of missile after perforation of steel barrier of thickness (T), ft/sec

V_s = striking velocity of missile normal to the target surface, ft/sec

V_p = velocity required to just perforate a barrier, ft/sec

V_p is calculated from Section 3.5.3.1.2.

Sufficient concrete thickness is provided so as to have residual velocity (V_r) of zero.

3.5.3.2 Overall Damage Prediction.

3.5.3.2.1 Impactive Load Analysis: Two techniques were used to determine the effect of impactive missile loads on a structure. In both methods, the missile impact load is expressed in the form of an equivalent static load resistance function, or load capacity, which the target structure must develop. The application of these methods depends on the nature of the impact. The energy method is applied to cases when the missile is small and fast and the penetration exceeds 15 percent of the target thickness, while the momentum method is used to analyze the impact effect of slow-moving large missiles. The above two methods are described in References 3.5-16, 3.5-17, 3.5-19, and 3.5-23. Ductility factors for these analyses are given in Table 3.5-13.

Structural integrity need not be considered for the immediate impact area within a circle having a diameter equal to the mean diameter of the impacting missile. Stability of the structure does not present a problem when ductility factors conform to those given in Tables 3.5-13. The yield displacement values for structural elements are shown in Tables 3.5-14 and 3.5-15. The maximum structural displacement has been estimated by multiplying corresponding values with the ductility factor used in the design.

3.5.3.2.2 Design of Concrete Barriers:

1. Impact Away from Supports

When the missile impacts in the vicinity of the middle of a two-way slab, the analytical approach as indicated in Table 3.5-15 is utilized. The resistance and yield displacement values are calculated in accordance with the boundary conditions and log/short sides ratio of the two-way slab. The ductility factors are used as shown in Table 3.5-13.

In the case where the missile strikes a beam, conventional analysis is performed as shown in Table 3.5-14. Limiting deflections and the corresponding effects to the ductility factors are shown in Table 3.5-13.

The transmission of the effects of these local loadings throughout the rest of the structure has been treated on an elastic basis in accordance with the acceptance criteria presented in Section 3.8.1.5 through 3.8.5.5.

2. Impact at or Adjacent to Supports

The local damage criteria of no spalling yields panel thicknesses sufficient to preclude overall structural damage.

3.5.3.2.3 Design of Steel Barriers:

1. Effects of Impact Away from a Support

In the analysis of impact effects on steel plate barriers for missile hits in the vicinity of the center of the plate, the resistance function specified in Table 3.5-15 was used in conjunction with the allowable ductility factors in Table 3.5-13.

2. Effects of Impact in the Vicinity of a Support

For impact effects in the vicinity of a support, it was sufficient that the Stanford penetration formula be satisfied. This automatically satisfied the possibility of punching shear.

REFERENCES

Section 3.5:

3.5-1 Not used.

3.5-2 Not used.

3.5-3 Not used.

3.5-4 Not used.

3.5-5 Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles", (March 1976).

3.5-6 Recht, R. F., and T. W. Ipson, "Ballistic Perforation Dynamics", Journal of Applied Mechanics, Transactions of ASME, Vol. 30, Series E, No. 3 (September 1963).

3.5-7 Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants" (April 1974).

3.5-8 Regulatory Guide 1.117, "Tornado Design Classification" (June 1976).

3.5-9 Not used.

3.5-10 Lon, Tom, Houston Air Traffic Control Center, Federal Aviation Administration, personnal communication with Sam Hobbs, Brown & Root, Inc., July 7, 1977, B&R memorandum GM-21867.

3.5-11 National Transportation Safety Board, Table of "Accidents, Fatalities, Rates, U. S. General Aviation" (January 4, 1977).

3.5-12 National Transportation Safety Board, "Annual Review of Aircraft Accident Data, U. S. General Aviation, Calendar Year 1975", NTSB_ARG-77-1.

3.5-13 Amirikian, A., Design of Protective Structures, Bureau of Yards and Docks (1950).

3.5-14 Moore, C. V., "The Design of Barricades for Hazardous Pressure Systems", Nuclear Engineering and Design (1967).

3.5-15 Westinghouse Electric Corporation, "PLASAR" (December 1972).

REFERENCES (Continued)

Section 3.5:

3.5-16 Biggs, J. M., Introduction to Structural Dynamics, McGraw-Hill (1964).

3.5-17 Stevenson, J. D., "Simplified Procedures for Analysis and Design of Concrete Structures Subjected to Missile Loads with Particular Application Nuclear Power Plant Facilities."

3.5-18 Not used.

3.5-19 Williamson, R. A. and B. R. Alvy, "Impact Effect of Fragments Striking Structural Elements", NP-6515 (1957).

3.5-20 Not used.

3.5-21 Not used.

3.5-22 Not used.

3.5-23 Manual, S. A., "Impactive Dynamic Analysis", ASCE National Structural Engineering Meeting, Baltimore, Maryland, (April 19-23, 1971).

3.5-24 Not used.

3.5-25 Not used.

3.5-26 Not used.

3.5-27 Not used.

3.5-28 Rotz, J. V., "Results of Missile Impact Tests on Reinforced Concrete Panels", Vol. 1A, pp 720-738, Second Specialty Conference on Structural Design of Nuclear Power Plant Facilities, New Orleans, LA, December 1975.

3.5-29 Gwaltney, R.C., "Missile Generation and Protection on Light-Water-Cooled Power Reactor Plants", ORNL NSIC-22, Oak Ridge National Laboratory, Oak Ridge, TN, for the USAEC, September 1968.

3.5-30 Price, Arnold E., Houston Air Traffic Control Center, Federal Aviation Administration to Rick Donahue, Bechtel, February 2, 1983.

REFERENCES (Continued)

Section 3.5:

3.5-31 Federal Aviation Administration, FAA Statistical Handbook of Aviation, DOT/FAA-SNA-1981, 1981.

3.5-32 Not used.

3.5-33 Timoshenko, S., and S. Woinowsky-Krieger, “Theory of Plates and Shells”, McGraw-Hill, 1959.

3.5-34 Not used.

3.5-35 Not used.

3.5-36 HL&P Letter, ST-HL-AE-4183, dated October 28, 1992, “Revised Turbine System Maintenance Program”.

3.5-37 NRC Safety Evaluation Report Related to the Turbine Maintenance Program, dated May 13, 1993.

3.5-38 Siemens Power Generation Missile Report CT-27385 for BB380-13.9m^2 Low Pressure Turbine Upgrade, signed 05DEC2005.

TABLE 3.5-1

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC

CATEGORY I STRUCTURES TO BE PROTECTED

System, Component, or Structure	Location	External Missile Protection*	Internal Missile Protection*	UFSAR Reference
Structures				Section 3.8
Containment Building	N/A	A	N/A	
Mechanical-Electrical Auxiliaries Building, including control room	N/A	A	N/A	
Fuel Handling Building	N/A	A	N/A	
Diesel Generator Building	N/A	A	N/A	
Essential Cooling Water Intake and Discharge structures	N/A	A	N/A	
Essential Cooling Pond	N/A	A	N/A	
MSIV structure (IVC)	N/A	A	N/A	
FW valve structure (IVC)	N/A	A	N/A	
Class 1E Underground Electrical Raceway System	N/A	A	N/A	
Auxiliary Feedwater Storage Tank	N/A	A	N/A	
Containment Isolation Valves and Piping	RCB, IVC, FHB, MAB	B	D	Section 6.2.4

*See notes at the end of this table for code meanings.

TABLE 3.5-1 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES TO BE PROTECTED

System, Component, or Structure	Location	External Missile Protection*	Internal Missile Protection*	UFSAR Reference
Reactor Coolant System				Chapter 5
Reactor vessel & supports	RCB	B	B	
CRDM assembly	RCB	B	B	
Thermal barrier	RCB	B	D	
Steam generator & supports	RCB	B	D	
Reactor Coolant pumps & supports	RCB	B	D	
Pressurizer & supports	RCB	B	D	
Other RCS piping, supports valves & fittings required to maintain RC pressure boundary	RCB	B	D	
Reactor Vessel head vent system	RCB	B	D	
Reactor Head Degassing System	RCB/MAB	B	D	
Chemical and Volume Control System				Section 9.3.4.1
Regenerative HX	RCB	B	D	
Centrifugal charging pump	MAB	B	D	
Positive displacement pump	MAB	B	D	
Seal water injection filter	MAB	B	D	
Seal water return filter	MAB	B	D	
Boric acid transfer pump	MAB	B	D	

TABLE 3.5-1 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES TO BE PROTECTED

System, Component, or Structure	Location	External Missile Protection*	Internal Missile Protection*	UFSAR Reference
Chemical and Volume Control System (Cont'd)				
Boric acid filter	MAB	B	D	
RC purification pump	MAB	B	D	
Boric acid tanks	MAB	B	D	
Pulsation damper	MAB	B	D	
Piping and valves	RCB/MAB	B	C	
Emergency Core Cooling System				Section 6.3
Accumulators	RCB	B	D	
HHSI pumps	FHB	B	D	
LHSI pumps	FHB	B	D	
Piping and Valves	RCB/FHB	B	D	
Refueling Water Storage Tank (RWST)	MAB	B	D	
Residual Heat Removal System				Section 6.3.1
Residual heat removal pump	RCB	B	D	
Residual HX	RCB	B	D	
Piping, supports and valves	RCB	B	D	

TABLE 3.5-1 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES TO BE PROTECTED

System, Component, or Structure	Location	External Missile Protection*	Internal Missile Protection*	UFSAR Reference
Containment Spray System				Section 6.5.2
Containment spray pump	FHB	B	D	
Spray additive eductor	FHB	B	D	
Containment spray nozzle	FHB	B	D	
Piping and valves	RCB/FHB	B	D	
Spent fuel Pool Cooling and Cleanup System				Section 9.1.3
Piping, supports and valves	FHB/MAB	B	D	
Spent fuel pool HX	FHB	B	D	
Spent fuel pool cooling pump	FHB	B	D	
Refueling water purification pump	MAB	B	D	
Reactor Makeup Water System				Section 9.2.7
Reactor makeup water storage tank	MAB	B	D	
Reactor makeup water pumps	MAB	B	D	
Piping and valves	RCB/MAB	B	D	

TABLE 3.5-1 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES TO BE PROTECTED

System, Component, or Structure	Location	External Missile Protection*	Internal Missile Protection*	UFSAR Reference
Main Steam				Section 10.3
Those portions of the MS System including supports extending from and including the secondary side of the SGs up to and including the first restraint outside the valve cubicle and connected piping up to and including the first vlave that is either normally closed or capable of automatic closure during all modes of normal reactor operation.	RCB/IVC	B	D	
Steamline to Turbine AFW pump	IVC	B	D	
Code safety valves	IVC	B	D	
Ms drain lines and valves to outside IVC wall	IVC	B	D	
Main steam isolation valves	IVC	B	D	
Steam generator PORVs	IVC	B	D	
Piping and valves	IVC	B	D	
Component Cooling Water System				Section 9.2.2
Heat exchangers	MAB	B	D	
Pumps	MAB	B	D	
Surge tank	MAB	B	D	
Piping and valves other than those required for isolation	RCB/MAB/FHB	B	D	
Vent and drain piping up to and including first isolation valve	RCB/MAB/FHB	B	D	

TABLE 3.5-1 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES TO BE PROTECTED

System, Component, or Structure	Location	External Missile Protection*	Internal Missile Protection*	UFSAR Reference
Essential Cooling Water System				Section 9.2.1.2
Essential cooling water pumps	ECWIS	B	D	
Strainers	ECWIS	B	D	
Screen Wash System	ECWIS	B	D	
Piping	MAB, DGB, ECWIS	B	D	
Valves	MAB, DGB, ECWIS	B	D	
Auxiliary Feedwater System				Section 10.4.9
Pumps	IVC	B	D	
Pump turbine	IVC	B	D	
AFW piping from AFST to AFW pumps	IVC	B	D	
AFW piping & valves from AFW pumps to SGs	RCB/IVC	B	D	
AFW pump test/recirc. lines inside IVC	IVC	B	D	

TABLE 3.5-1 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES TO BE PROTECTED

System, Component, or Structure	Location	External Missile Protection*	Internal Missile Protection*	UFSAR Reference
Feedwater System				Section 10.4.7
Those portions of the FW System extending from and including the secondary side of the SGs up to and including the first restraint outside the valve cubicle and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation.	RCB/IVC	B	D	
Sampling System				Section 9.3.2
Sample delay coil	RCB/MAB	B	D	
Piping and valves	RCB/MCB	B	D	
Steam Generator Blowdown System				Section 10.4.8
Piping from SG out to and including the isolation valves	RCB/IVC	B	D	
Diesel Generator Lube Oil System	DGB	B	D	
Diesel Generator Fuel Storage and Transfer System				Section 9.5.4
Diesel oil storage tanks	DGB	B	D	
Valves	DGB	B	D	
Piping except vent and fill piping downstream of last valve in the line	DGB	B	D	

TABLE 3.5-1 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES TO BE PROTECTED

System, Component, or Structure	Location	External Missile Protection*	Internal Missile Protection*	UFSAR Reference
Diesel Generator Cooling Water System	DGB	B	D	Section 9.5.5
Diesel Generator Air Starting System				Section 9.5.6
Air receivers	DGB	B	D	
Piping and valves	DGB	B	D	
Containment Hydrogen Monitoring System				Section 6.2.5
Hydrogen analyzer package	MAB	B	C	
Piping and valves inside the Containment to and including the analyzer package isolation valves	RCB/MAB	B	C	
Remaining piping and valves	MAB	B	C	
Heating, Ventilating, and Air Conditioning System				
1. Containment building HVAC RCFs including ductwork, MS isolation valve structure ductwork containment cubicle fans, exhaust fans, ductwork, dampers	RCB	B	D	Section 9.4.5
Containment purge isolation valves and penetrations	RCB/FHB MAB	B	D	

TABLE 3.5-1 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES TO BE PROTECTED

System, Component, or Structure	Location	External Missile Protection*	Internal Missile Protection*	UFSAR Reference
Heating, Ventilating, and Air Condition System (Cont'd)				
2. Mechanical Auxiliary Building HVAC				Section 9.4.3
Supplementary coolers subsystem	MAB	B	C	
3. Control Room and Electrical Auxiliary Building HVAC				Section 9.4.1
Air handling unit	EAB	B	D	
Fans, ductwork, and dampers	EAB	B	D	
Battery room exhaust fans	EAB	B	D	
Filters	EAB	B	D	
Chiller	MAB	B	D	
Chilled water pump, piping etc	MAB	B	D	
4. Fuel Handling Building HVAC				Section 9.4.2
Main exhaust fans, exhaust booster fans ductwork, and dampers	FHB	B	D	
Exhaust filters	FHB	B	D	
Supplementary coolers	FHB	B	D	
5. Diesel Generator Building emergency HVAC fans, ductwork, and dampers	DGB	B	D	Section 9.4.6

TABLE 3.5-1 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES TO BE PROTECTED

System, Component, or Structure	Location	External Missile Protection*	Internal Missile Protection*	UFSAR Reference
Heating, Ventilating, and Air Conditioning System (Cont'd)				
6. Essential cooling water pump room HVAC fans, ductwork, and dampers	ECWIS	B	D	Section 9.4.7
Class 1E Electrical System Components				Chapter 8
4,160/480 V switchgear (ESF buses)	EAB	B	D	
4,160/480 V transformers (ESF load centers)	EAB	B	D	
480/120/208Y V transformers (control room and ESF area lighting)	EAB	B	D	
480 V switchgear (ESF load centers)	EAB	B	D	
480 V motor control and MCCs (ESF MCCs)	EAB	B	D	
125 V station batteries and racks (control and vital instrumentation power supplies)	EAB	B	D	
480 vac/125 vdc battery chargers (For vital dc bus)	EAB	B	D	
115 vdc panels (vital dc power distribution)	EAB	B	D	
Voltage regulators (backup for instrumentation inverters)	EAB	B	D	

TABLE 3.5-1 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES TO BE PROTECTED

System, Component, or Structure	Location	External Missile Protection*	Internal Missile Protection*	UFSAR Reference
Class 1E Electrical System Components (Cont'd)				
120 vac instrument bus panels (vital instrumentation ac power distribution)	EAB	B	D	
Containment penetration assemblies	EAB/RCB	B	D	
Main control board	EAB	B	D	
ESF load sequencer	EAB	B	D	
Diesel generator and accessories	DGB	B	D	
Diesel generator control panels	O	B	D	
Relay boards and racks	EAB	B	D	
Wire and cable raceway system	I/O	B	C	
Underground electrical duct bank system	O	B	D	
Cable system (power, control, and instrumentation)	I/O	B	C	
Electrical supports	I/O	B	D	
Motors (1E)	I/O	B	D	
Valve operators	I/O	B	C	

TABLE 3.5-1 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC
CATEGORY I STRUCTURES TO BE PROTECTED

System, Component, or Structure	Location	External Missile Protection*	Internal Missile Protection*	UFSAR Reference
Instrumentation and Control System Components				Chapter 7
Radiation monitoring system (safety-related components)	I/O	B	C	
Reactor Trip System	I/O	B	D	
Engineered Safety Feature (ESF) Actuation System	EAB	B	D	
Systems required for safe shutdown	RCB/MAB FHB/TGB	B B	C D	
Post accident monitoring system	O	B	C	
Safety-related instruments, tubing, and fittings	I/O	B	C	
Safety-related process instruments	I/O	B	C	
Fuel Handling System				Section 9.1
Fuel transfer tube and flange	RCB/FHB	B	D	
Spent fuel racks	FHB	B	D	
Incore Instrumentation				Chapter 7
Seal table assembly	RCB	B	D	
Flux thimble tubing	RCB	B	D	
Flux thimble fittings	RCB	B	D	
Flux guide tubing	RCB	B	D	

TABLE 3.5-1 (Continued)

SAFETY CLASS SYSTEMS AND COMPONENTS AND SEISMIC CATEGORY I STRUCTURES TO BE PROTECTED

Location

I	- Inside Containment
O	- Outside Containment
N/A	- Not applicable
RCB	- Reactor Containment Building
FHB	- Fuel Handling Building
MAB	- Mechanical Auxiliary Building
EAB	- Electrical Auxiliary Building
DGB	- Diesel Generator Building
IVC	- Isolation Valve Cubicle
TGB	- Turbine Building
ECWIS	- Essential Cooling Water Intake Structure

External Missile Protection

A	- Designed to withstand the impact of an external missile
B	- Housed in a structure designed to withstand the impact of an external missile

Internal Missile Protection

A	- Designed to withstand the impact of internal missiles which might strike the component
B	- Protected from the impact of internal missiles by shield walls or the equivalent
C	- Protection not required due to component redundancy or not required to function to mitigate the consequences of the missile or enable safe shutdown
D	- Protection not required because no missiles strike the component
N/A	- Not applicable

TABLE 3.5-2

BARRIERS DESIGNED FOR MISSILES

Structure	Protection Afforded	Missile Type
Reactor vessel biological shield	The reactor vessel is protected from missiles originating within the SG primary shield wall.	Internal missiles resulting from pressurized components or rotating equipment
Steam generator secondary shield wall	The Containment and equipment located between the SG primary shield wall and the Containment are protected from missiles generated within the SG primary shield wall.	Internal missiles resulting from pressurized components or rotating equipment
Control rod drive mechanism missile shield	The missiles shield prevents the ejection into the Containment of the worst postulated missile from the head area.	Internal missiles resulting from pressurized components
Reactor Containment Building	The Containment will be designed to prevent missiles from damaging the liner.	External and internal missiles
Mechanical-Electrical Auxiliaries Building exterior walls and roof	Equipment located within the MEAB is protected from external missiles.	External missiles
Fuel Handling Building exterior walls and roof	Equipment located within the FHB is protected from external missiles.	External missiles
Control room exterior walls and roof	Equipment located within the control room is protected from external missiles.	External missiles
Diesel Generator Building exterior walls and roof	Equipment located within the DGB is protected from external missiles.	External missiles
Auxiliary Feedwater storage tank	The AFST is protected from external missiles.	External missiles

TABLE 3.5-2 (Continued)

BARRIERS DESIGNED FOR MISSILES

Structure	Protection Afforded	Missile Type
Isolation Valve Cubicle (IVC) Walls	The Main Steam (MS) Safety Valves, Feedwater and MS Isolation Valves are protected from external missiles	External missiles
Essential Cooling Water Intake and Discharge Structures	Equipment located within the structures is protected from internal and external missiles.	Internal and external missiles

TABLE 3.5-3

SUMMARY OF CONTROL ROD DRIVE MECHANISM MISSILE ANALYSIS

Postulated Missiles	Weight (lb)	Thrust Area (in.^{2})	Effective Impact Area (in.^{2})	Impact Velocity (ft/sec)	Kinetic Energy (ft-lb)	Penetration[(1)] (in.)
Mechanism housing plug - Not applicable for replacement CRDMs	50	4.91	0.87	40	1,242	0.163
Drive shaft	165	2.40	3.56	100	25,620	0.773
Drive shaft latched to mechanism	1,610	12.57	1.37	12	3,600	0.265

CN-2994

1. Ballistic Research Laboratories (for steel)

HISTORICAL INFORMATION CN-2994

TABLE 3.5-4

CHARACTERISTICS OF OTHER MISSILES
POSTULATED WITHIN REACTOR CONTAINMENT

	Pressurizer Heaters
Weight	15 lb
Discharge area	0.80 in.^{2}
Thrust area	2.4 in.^{2}
Impact area	2.4 in.^{2}
$\frac{\text{Missile weight}}{\text{Impact area}}$	6.25 psi
Velocity	55 ft/sec

TABLE 3.5-9

CHARACTERISTICS OF TORNADO-GENERATED MISSILES

Missile	Length (ft)	Weight (lb)	Velocity (ft/sec)*
4″ x 12″ wood plank	12	200	420
3″-diameter schedule 40 steel pipe	10	78	210
1″-diameter steel rod (reinforcing bar)	3	8	310
6″-diameter schedule 40 steel pipe	15	285	210
12″-diameter schedule 40 steel pipe	15	743	210
13.5″-diameter wooden utility pole	35	1,490	210
Automobile (4′ x 5′ frontal area)	15	4,000	100

The first five missiles are considered at all altitudes and the last two missiles at altitudes up to 30 ft above grade levels (except the Cooling Reservoir embankment) within one-half mile of the safety-related structures. There are no utility poles atop the embankment within one-half mile of the safety-related structures. There is an access road on top of the embankment, but there will be limited traffic on the road and then only on rare occasions, consisting only of authorized vehicles being used during inspection or maintenance activities. No part of the embankment is closer to safety-related structures, systems, or components than 650 ft.

* Assuming a Region I tornado, as defined in Regulatory Guide 1.76 (April 1974).

TABLE 3.5-10

BARRIERS FOR TORNADO MISSILES

		Concrete Thickness (in.)			
Protected Systems and Components	Missile Barrier	Walls	Roof	Concrete Strength (psi)	Curing Time (Days)
NSSS equipment, containment piping, electrical, instrumentation, control systems and containment ESF actuation systems, safety injection system.	Containment Structure	48	36	5500	90
Control room and electrical, instrumentation, control and ventilation equipment in EAB	Electrical Auxiliaries Building	30	24	4000	28
Mechanical, electrical, instrumentation and control equipment in MAB	Mechanical Auxiliaries Building	30	24	4000	28
Essential cooling water pumps and pump motors	ECW Intakes Structure	24	24	4000	28
Spent fuel pool and safety related equipment in FHB	Fuel Handling Building Exterior Fuel Pool Walls	36 66	24 -	4000 4000	28 28
Diesel generators, diesel generator fuel oil systems, fuel storage tanks, pumps and motors	Diesel Generator Building	24	24	4000	28
Diesel generator combustion air and ventilation air inlet	Diesel Generator Building & Piping Geometry	24	24	4000	28

TABLE 3.5-10 (Continued)

BARRIERS FOR TORNADO MISSILES

		Concrete Thickness (in.)			
Protected Systems and Components	Missile Barrier	Walls	Roof	Concrete Strength (psi)	Curing Time (Days)
Main steam line isolation valves and auxiliary feedwater pumps	Containment Structure Wall	48	-	5500	90
	Isolation Valve Cubicle Wall	24(a)	(b)	4000	28
Auxiliary feedwater storage tank	Concrete Tank Walls and Roof	30(c)	30(c)	4000	28
Auxiliary feedwater lines and valves	Valve Pit	24	24(e)	4000	28
Essential cooling water system piping	Underground	NA	NA		
Class 1E outside electrical raceway system	Underground[d]	NA	NA		

a. Minimum thickness.
b. Roof is metal deck. Risk analysis for tornado missile strike yields probability of $<10^{-7}$.
c. Including 1/4 in. stainless steel plate liner.
d. Except for raceway system from the MEAB to the TGB where protection is provided by 7-inch-thick cover of 5500 psi (at 90 days) concrete. Missile penetration is less than 7 inches.
e. Hatch covers are 3/4-inch-thick steel plate.

TABLE 3.5-11

GENERAL AVIATION ACCIDENT RATE*

Year	Total Accidents	Thousands of Miles Flown
1972	4,256	3,317,100
1973	4,255	3,728,500
1974	4,425	4,042,700
1975	4,237	4,238,400
1976	4,567	4,296,400
TOTAL	21,740	19,623,100
	X 16%**	
ADJUSTED TOTAL	3,478	

FIVE YEAR AVERAGE RATE = 1.8 X 10^{-7} accidents/mile

* Based on January 4, 1977 National Transportation Safety Board table of "Accidents, Fatalities, Rates, U.S. General Aviation, 1966-1976" (Ref. 3.5-11).

** Based on fraction of accidents for climbing, normal cruise, and descending reported from the "Annual Review of Aircraft Accident Data, U.S. General Aviation, Calendar Year 1975", NTSB-ARG-77-1 (Ref. 3.5-12).

TABLE 3.5-12

AIRCRAFT ACCIDENTS, FATALITIES AND ACCIDENT RATES—
U.S. GENERAL AVIATION FLYING: 1972-1981[C]

YEAR	ACCIDENTS TOTAL	ACCIDENTS FATAL	FATALITIES	AIRCRAFT HOURS FLOWN (X 1000)	ACCIDENT RATES 100,000 AIRCRAFT HOURS TOTAL	ACCIDENT RATES 100,000 AIRCRAFT HOURS FATAL
1972R	*4,109	*653(a)	1,305(b)	24,419	16.8	2.67
1973R	*4,090	*679(a)	1,299	26,908	15.2	2.52
1974R	*4,234	*689(a)	1,327	27,774	15.2	2.47
1975R	*4,034	*638(a)	1,247	28,336	14.2	2.24
1976R	*4,005	*648(a)	1,187	29,975	13.3	2.15
1977R	*4,069	*658(a)	1,281	31,585	12.9	2.08
1978R	*4,223	*723(a)	1,563(b)	34,985	12.1	2.07
1979R	*3,800	*629(a)	1,219	38,767	9.8	1.62
1980R	*3,599	*629(a)	1,264	37,480	9.6	1.68
1981P	3,634	662	1,265	36,280	10.0	1.82

a. Suicide/Sabotage Accidents are included in all computations except for rates (1972-3, 1973-2, 1974-2, 1975-2, 1976-4, 1977-1, 1978-2, 1979-0).

b. Includes air carrier fatalities (1972-5, 1978-142) when in collision with General Aviation Aircraft.

c. Reference 3.5-31.

SOURCE: National Transportation Safety Board.

P – Preliminary.
R – Revised.

* As of 1981 General Aviation no longer includes air taxi (commuter air carrier and on-demand air taxi) accidents. The number of total accidents, fatal accidents, fatalities, and aircraft hours flown and accident rates for the years 1972-1980 have been adjusted to accommodate the exclusion of air taxi accidents and air taxi hours flown.

TABLE 3.5-13

DUCTILITY FACTORS

1. Reinforced concrete beams and slabs controls design with on-way reinforcement

$$\mu = \frac{0.05}{p - p'} ; \ \mu \le 10.0 \quad \text{where} \quad p = \frac{A_s}{bd}$$

2. Reinforced concrete and slabs controls design with two-way reinforcement(1)

$$\mu = \frac{0.05}{p - p'} ; \ \mu \le 10.0 \quad \text{where} \quad p' = \frac{A'_s}{bd}$$

3. Concrete beams and slab in region controlled by shear

   a) Shear carried by concrete and stirrups — $\mu = 1.3$

   b) Shear carried completely by stirrups — $\mu = 3.0$

   c) Shear carried by concrete alone — $\mu = 1.0$

4. Concrete columns and walls (compression members) — $\mu = 1.3$

5. Structural steel tension members(2)

$$\mu = 0.5 \frac{\varepsilon_u}{\varepsilon_y}$$

6. Structural steel flexural members

   a) Open sections (I, WF, T, etc. Members proportioned to preclude lateral and local plastic buckling) $\mu \le 10.0$

A_s = area of tension reinforcement

A'_s = area of compressive reinforcement

b = width of section

d = effective depth of section

p = percentage tensile reinforcement

p' = percentage compression reinforcement

ε_u = uniform ultimate strain of material

ε_y = strain at yield of material

TABLE 3.5-13 (Continued)

DUCTILITY FACTORS

b) Closed sections (pipe box, etc.)	$\mu \leq 10.0$
c) Members where shear governs design	$\mu \leq 6.0$
7. Structural steel columns	$\mu = 1.3\ 1/r \leq 20$ $\mu = 1.0\ 1/r > 20$

l = effective length of column

r = radius of gyration (see AISC-1969 Specifications)

---

1. Ductility ratio up to 30 can be used provided the angular rotation per following equation is satisfied.

$$r_\theta = .0065 \frac{d}{c} \leq 0.07$$

where:

r_θ = hinge rotation (radians)

d = distance from compression face to centroid of tensile steel reinforcement (in.)

c = distance from compression face to the neutral axis at ultimate strength (in.)

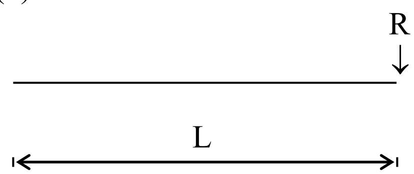
2. In lieu of actual test values, ε_u may be taken as the strain corresponding to 50% of ASTM specified minimum elongation.

TABLE 3.5-14

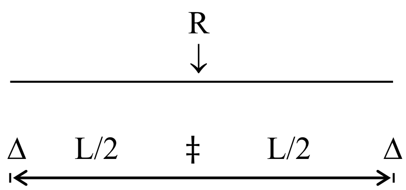
RESISTANCE/YIELD DISPLACEMENT VALUES FOR BEAMS

DESCRIPTION	RESISTANCE	YIELD
(1) CANTILEVER	$R = \frac{M_u}{L}$	$X_y = \frac{RL^3}{3EI}$
(2) SIMPLY SUPPORTED	$R = \frac{4M_u}{L}$	$X_y = \frac{RL^3}{48EI}$
(3) FIXED SUPPORTS	$R = \frac{8M_u}{L}$	$X_y = \frac{RL^3}{192EI}$
(4) MULTI-SPAN	$R = \frac{8M_u}{L}$	$X_y = \frac{0.011RL^3}{EI}$

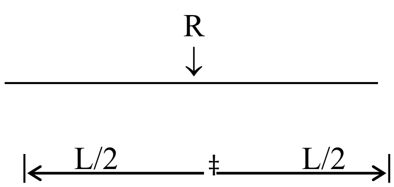
(1) CANTILEVER



(2) SIMPLY SUPPORTED



(3) FIXED SUPPORTS



(4) MULTI-SPAN

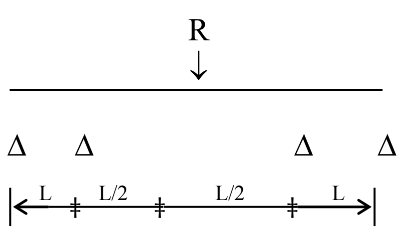
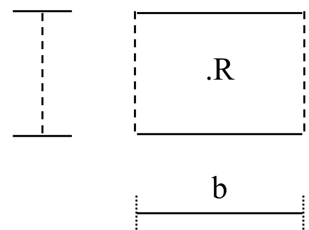


TABLE 3.5-15

RESISTANCE/YIELD DISPLACEMENT VALUES FOR RECTANGULAR SLABS*

Yield Displacement at Center

$$X_y = \frac{\alpha R a^2}{EI} (1 - \mu^2)$$



where:

R	=	Yield resistance
a	=	Short side of slab
b	=	Long side of slab
μ	=	Possion's ratio
E	=	Modulus of elasticity
I	=	Moment of inertia per unit width
M_u	=	Ultimate moment capacity per unit width

(1) Simple supported on all four sides with load at center

Resistance R = 2πMu

b/a	1.0	1.1	1.2	1.4	1.6	1.8	2.0	3.0	∞
α	.01160	.01265	.01353	.01484	.01570	.01620	.01651	.01690	.01690

(2) Fixed supports on all four sides with load at center

Resistance R = 4πMu

b/a	1.0	1.2	1.4	1.6	1.8	2.0	∞
α	.00560	.00647	.00691	.00712	0.00720	.00722	.00725

* Source: Timoshenko, S., and S. Woinowsky-Krieger, "Theory of Plates and Shells", McGraw-Hill (1959) (Ref. 3.5-33).

TABLE 3.5-16

ROTATING EQUIPMENT MISSILE SOURCES

OUTSIDE CONTAINMENT

Equipment Identification	Location	Casing Perforation (Yes/No)	Calculated Thickness to Prevent Concrete Spalling	Remarks
ECW Pumps	ECW Intake Structure	Yes - Assumed	-	(a)
Centrifugal Charging Pumps 1A, 1B	MAB	No	-	-
Boric Acid Transfer Pumps	MAB	No	-	-
CCW Pumps	MAB	No	-	-
Reactor Makeup Water Pumps	MAB	No	-	-
ECW Screen Wash Booster Pumps	ECWIS	No	-	-
Essential Chilled Water Pumps	MAB	No	-	-
Feedwater Isolation Valves Hydraulic Pump Modules	IVC	Yes - Assumed	-	(a)
Refueling Water Purification Pump	MAB	No	-	-
MAB Chilled Water Pumps	MAB	No	-	-
Waste Evaporator Recirculation Pumps	MAB	Yes - Assumed	-	(a)
Low Activity Spent Resin Sluice Pump	MAB	Yes - Assumed	-	(a)
Waste Concentrates Transfer Pump	MAB	Yes - Assumed	-	(a)
Condensate Polishing Waste Collection Tank Transfer Pump	MAB	Yes - Assumed	-	(a)
LWPS Evaporator Distillate Pump	MAB	Yes – Assumed	-	(a)

TABLE 3.5-16 (Continued)

ROTATING EQUIPMENT MISSILE SOURCES
OUTSIDE CONTAINMENT

Equipment Identification	Location	Casing Perforation (Yes/No)	Calculated Thickness to Prevent Concrete Spalling	Remarks
BRS Evaporator Pumps	MAB	Yes – Assumed	-	(a)
LWPS Seal Water Pumps	MAB	Yes – Assumed	-	(a)
Resin Dewatering Pump	MAB	Yes – Assumed	-	(a)
Spent Resin Transfer Pump	MAB	Yes – Assumed	-	(a)
LWPS Evaporator Condensate Return Pump	MAB	Yes – Assumed	-	(a)
Spent Fuel Cask Pool Pump	FHB	Yes – Assumed	-	(a)
Waste Holdup Tank Pump	MAB	Yes – Assumed	-	(a)
LWPS Surge Tank Pumps	FHB	Yes – Assumed	-	(a)
Waste Condensate Tank Pumps	MAB	Yes – Assumed	-	(a)
Waste Monitor Tank Pumps	MAB	Yes – Assumed	-	(a)
Floor Drain Tank Pumps	MAB	Yes – Assumed	-	(a)
Spent Fuel Pool Cooling Pumps 1A, 1B	FHB	No	-	-
Reactor Coolant Purification Pump	MAB	No	-	-
BTRS Chiller Pumps 1A, 1B	MAB	No	-	-
Spent Fuel Pool Skimmer Pump	FHB	No	-	-
RCFC Chilled Water Pump	MAB	No	-	-
LWPS Auxiliary Feed Pump	MAB	Yes – Assumed	-	(a)
Laundry and Hot Shower Tank Pump	MAB	Yes – Assumed	-	(a)

TABLE 3.5-16 (Continued)

ROTATING EQUIPMENT MISSILE SOURCES
OUTSIDE CONTAINMENT

Equipment Identification	Location	Casing Perforation (Yes/No)	Calculated Thickness to Prevent Concrete Spalling	Remarks
BRS Evaporator Feed Pumps	MAB	Yes – Assumed	-	(a)
Auxiliary Feedwater Pumps	IVC	No	-	-
BRS Condensate Return Pumps	MAB	Yes - Assumed	-	(d)
TSC Chilled Water Pumps	EAB	No	-	-
FHB Main Exhaust Fans	FHB	No	-	(c)
MAB Main Supply Fans	MAB	No	-	(c)
MAB Supplemental Exhaust Fans	MAB	No	-	(c)
Penetration Space Exhaust Fans	MAB	No	-	(c)
Tendon Gallery Fans	Tendon Gallery	No	-	-
MAB Main Exhaust Fan	MAB	Yes	<2 inches	(b)
RCB Normal Purge Supply Fan	MAB	No	-	-
RCB Normal Purge Exhaust Fan	MAB	No	-	-
Electrical Penetration Area AHU Fans	EAB	No	-	(c)
EAB Air Handling Unit Fans	EAB	No	-	-
Low Pressure Breathing Air Compressor	MAB	No	-	-
BTRS Chiller Compressor	MAB	No	-	-

TABLE 3.5-16 (Continued)

ROTATING EQUIPMENT MISSILE SOURCES
OUTSIDE CONTAINMENT

Equipment Identification	Location	Casing Perforation (Yes/No)	Calculated Thickness to Prevent Concrete Spalling	Remarks
RCB Supplementary Purge Supply Fans	MAB	No	-	(c)
RCB Supplementary Purge Exhaust Fans	MAB	No	-	(c)
CCW Pump Supplementary Cooler AHU Fans	MAB	No	-	-
Centrifugal Charging Pump Pump Supplementary Cooler AHU Fans	MAB	No	-	-
PD Charging Pump Supplementary Cooler AHU Fan	MAB	No	-	-
MAB Supplemental Fan Coil Units Fans	MAB	No	-	(c)
EAB Return Fans	EAB	No	-	-
EAB AHU Supply Fans	EAB	No	-	-
FHB Exhaust Booster Fans	FHB	No	-	-
FHB Supply Fans	FHB	No	-	-
DGB Oil Tank Room Exhaust Fan	DGB	No	-	(c)
Control Room Kitchen and Toilet Exhaust Fan	EAB	No	-	-
PASS Facility AHU Fan	FHB	No	-	(c)
FHB Elevator Exhaust Fans	FHB	No	-	-
Computer Room AHU Fans	EAB	No	-	(c)

TABLE 3.5-16 (Continued)

ROTATING EQUIPMENT MISSILE SOURCES
OUTSIDE CONTAINMENT

Equipment Identification	Location	Casing Perforation (Yes/No)	Calculated Thickness to Prevent Concrete Spalling	Remarks
TSC Computer Room AHU Fans	EAB	Yes - Assumed	-	(a)
Control Room AHU Supply Fans	EAB	No	-	(c)
TSC Makeup Air Fan	EAB	Yes – Assumed	-	(a)
TSC Supply Fans	EAB	Yes – Assumed	-	(a)
TSC Return Fans	EAB	Yes – Assumed	-	(a)
TSC Exhaust Fans	EAB	Yes – Assumed	-	(a)
TSC HVAC Equipment Room Exhaust Fan	EAB	Yes – Assumed	-	(a)
TSC Chiller	EAB	Yes – Assumed	-	(a)
Locker Room/Office Supply Fan	MAB	Yes – Assumed	-	(a)
Radwaste Counting Room AHU Fan	MAB	Yes - Assumed	-	(a)
Spent Fuel Pool Cooling Pump Rooms AHU Fans	FHB	No	-	-

a. Potential missiles from this source are separated from other essential systems by adequate barriers.
b. Missiles from this source which might penetrate the housing or casing will not interact with any equipment necessary to support safe shutdown or prevent uncontrolled releases of radioactivity.
c. Missiles which might exit the scroll of this centrifugal fan will not cause interactions which might prevent safe shutdown of the plant or result in an uncontrolled release of radioactivity.
d. Missiles which might penetrate the casing of this component will not cause interactions which could prevent safe shutdown of the plant or result in an uncontrolled release of radioactivity.

TABLE 3.5-17

ROTATING EQUIPMENT MISSILE SOURCES

OUTSIDE CONTAINMENT

Equipment Identification	Location	Casing Perforation (Yes/No)	Calculated Thickness to Prevent Concrete Spalling	Remarks
RHR Pumps	El. (-)4 ft-6 in.	No	-	-
Reactor Coolant Drain Tank Pumps	El. (-)9 ft-1 in.	No	-	-
RCFC Supply Fans	El. (-)2 ft-0 in.	No	-	-
Containment Cubicle Exhaust Fans	El. 68 ft-0 in.	No	-	-
Reactor Cavity Vent Fans	El. (-) 11 ft-3 in.	Yes	<2 inches	(a)
Containment Carbon Unit Supply Fans	El. 52 ft	No	<12 inches	(b)
Reactor Supports Exhaust Fans	El. 11 ft	No	-	-
CRDM Cooling Fans	El. 65 ft	No	-	-
RCB Elevator Vent Fan	El. 93 ft-8 in.	No	-	-

a. Missiles from this source which might penetrate the housing or casing will not interact with any equipment necessary to support safe shutdown or prevent uncontrolled releases of radioactivity.

b. The containment carbon unit fans have a wire screen over the discharge which may not stop a postulated missile from leaving the scroll. The containment liner might be impacted by such a missile from two of the fans, but the liner would not be perforated. The loss of other equipment which might be damaged by a missile impact would not prevent safe shutdown of the plant nor result in uncontrolled release of radioactivity.

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

Pipe failure protection is provided in accordance with the requirements of 10CFR50, Appendix A, General Design Criterion (GDC) 4.

In the event of a high- or moderate-energy pipe failure within the plant, adequate protection is provided to ensure that the essential structures, systems, or components are not adversely impacted by the effects of postulated piping failure. Essential systems and components are those required to shut down the reactor and mitigate the consequences of the postulated piping failure.

Appendix 3.6.B provides several examples of evaluations of the effects of postulated high energy pipe failures within the plant. The following sections provide the basis for selection of the pipe failures, the determination of resultant effects, and details of protection requirements.

3.6.1 Postulated Piping Failures in Fluid Systems Inside and Outside Containment

Table 3.6.1-1 provides a matrix of plant systems that indicates their classification: high-energy, moderate-energy, essential, or nonessential. Selection of pipe failure locations and evaluation of the consequences on nearby essential systems, components, and structures are presented in Section 3.6.2 and are in accordance with the requirements of 10CFR50, Appendix A, GDC 4. Selections and evaluations are in accordance with the guidance of Nuclear Regulatory Commission (NRC) Branch Technical Positions (BTP) ASB3-1 and MEB 3-1. The original design basis postulated pipe break locations in the reactor coolant loop (RCL) are described in Reference 3.6-1. A detailed fracture mechanics evaluation, as described in Reference 3.6-14, demonstrates that the probability of rupturing RCL piping is extremely low under design basis conditions. Therefore, postulated RCL ruptures and the following associated dynamic effects are not included in the design basis: missile generation, pipe whip, break reaction forces, jet impingement forces, decompression waves within the ruptured pipe, and pressurization in cavities, subcompartments and compartments. In addition, the dynamic effects from postulated pipe breaks have been eliminated from the structural design basis of the pressurizer surge line and the safety injection system accumulator lines. The elimination of the branch line breaks is based on the leak before break (LBB) analysis results presented in References 3.6-21 through 3.6-29, and 3.6-36. To provide high margins of safety required by GDC 4, the non-mechanistic pipe rupture design basis is maintained for containment design and ECCS analyses, and the postulated pipe ruptures are retained for electrical and mechanical equipment environmental qualification.

3.6.1.1 Design Bases. The following design bases relate to the evaluation of the effects of the pipe failures determined in Section 3.6.2.

1. The selection of the failure type is based on whether the system is high- or moderate-energy during normal operating conditions of the system.

   High-energy piping includes those systems or portions of systems in which the maximum normal operating temperature exceeds 200°F or the maximum normal operating pressure exceeds 275 psig.

Piping systems or portions of systems pressurized above atmospheric pressure during normal plant conditions and not identified as high-energy are considered moderate-energy.

Piping systems that exceed 200°F or 275 psig for about 2 percent or less of the time the system is in operation or that experience high-energy pressures or temperatures for less than 1 percent of the plant operation time are considered moderate-energy.

2. The following assumptions are used to determine the thermodynamic state in the piping system for the calculation of fluid reaction forces:

   a. For those portions of piping systems normally pressurized during operation at power, the thermodynamic state in the pipe and associated reservoirs are those of full-power (100 percent) operation.

   b. For those portions of piping systems only pressurized during other normal plant conditions (e.g., startup, hot standby, reactor cooldown), the thermodynamic state and associated operating condition is determined as the mode giving the highest enthalpy.

3. Moderate-energy pipe cracks are evaluated for spray wetting, flooding, and other environmental effects.

4. Where postulated, each longitudinal or circumferential break in high-energy fluid system piping or leakage crack in moderate-energy fluid system piping is considered separately as a single initiating event occurring during normal plant conditions.

5. Offsite power is assumed to be unavailable if a trip of the turbine-generator (TG) system or trip of the reactor is a direct consequence of the postulated piping failure.

6. A single active component failure is assumed in systems used to mitigate the consequences of the postulated piping failure or to safely shut down the reactor, except as noted in item 7, below. The single active component failure is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure, such as unit trip and loss of offsite power (LOOP).

7. When the postulated piping failure occurs in one of two or more redundant trains of a dual-purpose, moderate-energy essential system, single failures of components in other trains are not assumed, because the system is designed to seismic Category I standards; powered from both offsite and onsite sources; and constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety systems.

   Failures are not assumed in a system or component which is normally operating at the time of break initiation and which also functions (without change in state) to mitigate the break event, provided the system is designed to seismic Category I requirements and is qualified for the environment associated with the break event.

8. All available systems, including those actuated by operator actions, are employed to mitigate the consequences of a postulated piping failure to the extent clarified in the following paragraphs:

a. In determining the availability of the systems, account is taken of the postulated failure and its direct consequences, such as unit trip and Loop, and of the assumed single active component failure and its direct consequences. The feasibility of carrying out operator actions is determined on the basis of ample time and adequate access to equipment being available for he proposed actions. Although a postulated high/moderate-energy line failure outside the containment may ultimately require a cold shutdown, operation at hot standby is allowed in order for plant personnel to assess the situation and make repairs.

b. The use of nonseismic Category I systems in mitigating the consequence of postulated piping failure (other than a main steam system piping failure) outside the containment is clarified in the following paragraphs:

   1) For nonseismic Category I piping failures, it is assumed that a safe shutdown earthquake could be the cause of the failure. Therefore, only seismic Category I equipment can be used to mitigate the consequences of the failure and bring the plant to a safe shutdown.

   2) A postulated failure in seismically qualified portions of piping systems is not assumed to be seismically induced. Propagation of the failure to failures of nonseismically qualified equipment is not assumed. Nonseismic Category I equipment can be used to bring the plant to a safe shutdown following a postulated failure in seismically qualified piping, subject to power being available to operate such equipment and providing the equipment is qualified for the environment resulting from the piping failure.

9. A whipping pipe is not considered capable of rupturing impacted pipes of equal or greater nominal pipe diameter and equal or greater wall thickness.

   Unless shown otherwise by analysis, a whipping pipe is considered capable of developing a through-wall leakage crack in a pipe of larger nominal pipe size with thinner wall thickness.

   Impact against rigid steel electrical conduit, whose nominal pipe size and wall thickness are equal to or greater than those of the whipping pipe, is not assumed to damage the impacted conduit. If the conduit size is smaller than that of the whipping pipe, the conduit damage threshold is taken to be exceeded and cables within are assumed to fail.

10. Pipe whip is assumed to occur in the plane defined by the initial axis of the jet thrust force and a plastic hinge point.

   If unrestrained, a whipping pipe having a jet thrust force sufficient to form a plastic hinge is considered to rotate about the plastic hinge point. The whipping pipe will continue in motion until it is stopped by a structure or component of sufficient strength to withstand the loading imposed by the whipping pipe.

   In general, whipping ends from a pipe break are restrained so that plastic hinge formation is not allowed to occur. Where a plastic hinge could be formed, the effects are evaluated. Pipe whip restraints are provided wherever postulated pipe breaks could impair the ability of any essential system or component to perform its intended safety functions.

11. The calculation of thrust and jet impingement forces considers any line restrictions (e.g., flow limiter) between the pressure source and break location and the absence of energy reservoirs, as applicable.

12. Initial pipe break events are not assumed to occur in pump and valve bodies because of their greater wall thickness and their usual location in the low stress portions of the piping systems.

13. Where a system consisting of piping, restraints, and supporting structures is so complex that the assumption of planar motion is neither conservative nor realistic, the zone of whip influence is conservatively enlarged to a region approaching a sphere with a radius equal to the distance between the breakpoint and the first restraint. In lieu of this assumption a more detailed elastoplastic analysis is performed.

14. No loss of pressure boundary integrity is assumed from jet impingement, regardless of pressure, when the ruptured pipe has a diameter and wall thickness less than those of the impinged piping. For essential piping, jet impingement loads are evaluated regardless of the ratio of impinged and postulated broken pipe sizes.

15. Components impacted by jets from breaks in piping containing high pressure (870 to 2465 psia) steam or subcooled liquid that flashes at the break, such as piping connected to the steam generators or reactor coolant loops, shall be evaluated as follows:

    a. Unprotected components within 10 inside diameters of the broken pipe are assumed to fail. Specific jet loads are calculated and evaluated only when failure of the component, when combined with a single active failure, could adversely affect safe shutdown capability. These jet load calculations will be performed in accordance with Section 3.6.2.3.1.

    b. Unprotected components beyond 10 inside diameters of the broken pipe are considered undamaged by the jet without further analysis. The basis for this criteria is contained in Reference 3.6.13.

3.6.1.2 Description. Systems, components, and equipment required to perform the essential functions are reviewed to ensure conformance with the design bases and to determine their susceptibility to the failure effects. The break and crack locations are determined in accordance with Section 3.6.2 Figure 3.6.1-1 shows the high-energy pipe break locations, break types, and restraint locations.

A design comparison to NRC BTP ASB 3-1 and MEB 3-1 is provided in Tables 3.6.1-2 and 3.6.1-3.

Pressure response analyses are performed for subcompartments containing high-energy piping. For a detailed discussion of the pipe breaks selected and pressure results, refer to Section 6.2.1 for selected subcompartments inside the Containment and to Appendix 3.6.A for selected subcompartments outside the Containment. Effects of internal reactor pressure vessel asymmetric pressurization loads are addressed in Section 3.9.2. Asymmetric compartment pressurization loads inside Containment are addressed in Section 6.2.1. The analytical methods used for pressure response analysis are in accordance with Reference 3.6-2.

There are no high-energy lines in the proximity of the control room; therefore, there are no effects upon the habitability of the control room resulting from postulated pipe breaks. Further discussion of the control room habitability systems is provided in Section 6.4.

3.6.1.3 Safety Evaluation.

3.6.1.3.1 General: An analysis of postulated pipe failures is performed to determine the impact of such piping failures on those safety-related systems or components which are required to mitigate the consequences of the failure. By means of protective measures, such as separation, barriers, and pipe whip restraints, the effects of breaks and cracks are prevented from damaging essential items to an extent that would impair their essential function or necessary component operability. Typical measures used for protecting the essential systems, components, and equipment are outlined below and are discussed in detail in Section 3.6.2. The ability of specific safety-related systems to withstand a single active failure concurrent with the postulated event is discussed, as applicable. When the results of the pipe failure effects analysis show that the effects of a postulated pipe failure are isolated, physically remote, or restrained by protective measures from essential systems or components, no further dynamic hazards analysis is performed.

3.6.1.3.2 Protection Mechanisms: The plant layout arrangement is based on maximizing the physical separation of redundant or diverse safety-related components and systems from each other and from non safety-related items. Therefore, in the event a pipe failure occurs, there is a minimal effect on other essential systems or components required for safe shutdown of the plant or to mitigate the consequences of the failure.

The effects associated with a particular pipe failure must be mechanistically consistent with the failure. Thus, pipe dimensions, pipe layouts, material properties, and equipment arrangements are considered in defining the specific measures for protection against the consequences of postulated failures.

Protection against the dynamic effects of pipe failures is provided in the form of physical separation of systems and components, barriers, equipment shields, and pipe whip restraints. The precise method chosen depends largely upon considerations such as accessibility and maintenance.

1. Separation

   The plant arrangement provides separation, to the extent practicable, between redundant safety systems (including their appurtenances) to prevent loss of safety function as a result of hazards for which the system is required to be functional. Separation between redundant safety systems, with their related appurtenances, therefore, is the basic protective measure incorporated in the design to protect against the dynamic effects of postulated pipe failures.

   In general, layout of the facility follows a multi-step process to ensure adequate separation:

   a. Safety-related systems are located remotely from high-energy piping, where practicable.

b. Redundant safety systems are located in separate compartments.

c. As necessary, specific components are enclosed to retain the redundancy required for those systems that must function as a consequence of specific piping failure.

d. Drainage systems are reviewed to ensure their adequacy for flooding control.

2. Barriers and Shields

Protection requirements are met through the protection afforded by walls, floors, columns, abutments, and foundations. Where adequate protection does not already exist as a result of separation, additional barriers, deflectors, or shields are provided to meet the functional protection requirements.

Inside the containment, the secondary shield wall serves as a barrier between the RCLs and the containment liner. In addition, the refueling cavity walls, operating floor, and secondary shield walls minimize the possibility of an accident which may occur in any one reactor coolant loop affecting another loop or the containment liner. Those portions of the steam and feedwater (FW) lines located within the Containment are routed in such a manner that possible interaction between these lines and the reactor coolant piping is minimized. The barriers withstand loadings caused by jet forces and pipe whip impact forces.

Further discussion of barriers and shields is provided in Section 3.6.2.4.

3. Piping Restraint Protection

Measures for protection against pipe whip are provided where the unrestrained movement of the ruptured pipe could cause damage at an unacceptable level to any structure, system, or component required to meet the criteria outlined in Section 3.6.1.1.

The design criteria for and description of pipe whip restraints are given in Section 3.6.2.3.

3.6.1.3.3 Specific Protection Considerations:

1. Except for a main steam system piping failure, nonessential systems, structures and components are used to mitigate the consequences of a postulated pipe rupture (See Section 3.6.1.1.8).

2. High-energy containment penetrations are subject to special protection mechanisms. As discussed in Section 3.6.2.1.1.5, isolation restraints are located as close as practicable to the Containment isolation restraints are located as close as practicable to the Containment isolation valves associated with these penetrations. These restraints are provided, as appropriate, to maintain the operability of the isolation valves and the integrity of the penetration due to a break either upstream or downstream or the respective isolation restraints.

3. Safety-related instrumentation that is required to mitigate the effects of the pipe rupture is protected.

4. High-energy fluid system pipe whip restraints and protective measures are designed so that a postulated break in one pipe cannot, in turn, lead to a rupture of other essential pipes or components.

5. For any postulated loss-of-coolant accident (LOCA), the structural and leaktight integrity of the Containment is maintained.

6. The escape of steam, water, combustible or corrosive fluids, gases, and heat in the event of a pipe rupture will not preclude:

    a. Subsequent access to any areas, as required, to cope with the postulated pipe rupture.

    b. Habitability of the control room.

    c. The ability of essential instrumentation, electric power supplies, components and controls to perform their safety functions to the extent necessary to meet the criteria outlined in Section 3.6.1.1.

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

This section describes the design bases for locating postulated breaks and cracks in high- and moderate-energy piping systems inside and outside of the Containment; the methodology used to define the jet thrust reaction at the break location; the methodology used to define the jet impingement loading on adjacent essential structures, systems or components; pipe whip restraint design; and the protective assembly design.

3.6.2.1 Criteria Used to Define High/Moderate-Energy Break/Crack Locations and Configurations. NRC MEB 3-1, Reference 3.6-3, is used as the basis of the criteria for the postulation of high-energy pipe breaks. Specific moderate-energy pipe crack locations are not ascertained; and, therefore, they are assumed to occur as described in Section 3.6.2.1.2.

A postulated high-energy pipe break is defined as a sudden, gross failure of the pressure boundary of a pipe either in the form of a complete circumferential severance (i.e., a guillotine break) or as a sudden longitudinal, uncontrolled crack. For moderate-energy fluid systems, pipe failures are confined to postulation of controlled cracks in piping. The effects of these cracks in moderate energy fluid systems on the safety-related equipment are analyzed for flooding and wetting only. These cracks do not result in jet impingement or whipping of the cracked piping.

Breaks as stated above are postulated in each pipe and branch run adjacent to a protective structure or compartment containing essential systems and components.

Piping is considered adjacent to a protective structure or compartment containing essential systems and components required for safe shutdown if the distance between the piping and structure is insufficient to preclude impairment of the structure’s integrity from the effects of a postulated piping failure, assuming that the piping is unrestrained.

3.6.2.1.1 High-Energy Break Locations: With the exception of those portions of the piping identified in Section 3.6.2.1.1.5, breaks are postulated in high-energy piping at the following locations:

1. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1 – Class 1 Piping.

a. Pipe failure protection is provided in accordance with the requirements of 10 CFR 50, Appendix A, GDC 4. The original design postulated pipe break locations in the reactor coolant loop are described in Reference 3.6-1. In accordance with the provisions of GDC 4 (as revised per 52 FR 41294, October 27, 1987), the dynamic effects associated with postulated pipe breaks can be eliminated from the structural design basis if it is demonstrated that the probability of pipe rupture is extremely low. The dynamic effects that can be eliminated include missile generation, pipe whip, break reactor forces, jet impingement forces, decompression waves within the ruptured pipe, and pressurization in cavities, sub-compartments and compartments.

   Through the application of LBB technology, the dynamic effects from postulated breaks in the reactor coolant loop (primary) piping, the 16-inch pressurizer surge line, and the three SIS accumulator lines can be eliminated from structural design basis, based on the evaluations presented in References 3.6-14 and 3.6-21 through 3.6-29, and 3.6-36. The extent of application of LBB to the accumulator lines includes the 12-inch portions from the loop connections to the second check valve, and the connecting 8-inch and 10-inch lines to the first check valve. NRC approval of elimination of breaks in the Units 1 and 2 primary loop piping is given in Reference 3.6-30, and for the pressurizer surge line and SIS accumulator lines in References 3.6-31, 3.6-32, and 3.6-37. To provide the high margins of safety required by GDC-4, the non-mechanistic pipe rupture design basis is maintained for containment design and ECCS analysis, and the postulated pipe ruptures are retained for electrical and mechanical equipment environmental qualification.

b. For Class 1 piping not covered by exclusions noted in paragraph (a) above, pipe breaks are postulated to occur at the following locations in ASME Code Section III Class 1 piping runs or branch runs outside the RCL as follows:

   1) At terminal ends of the piping, including:

      a) Piping connected to structures, components, or anchors that act as essentially rigid restraints to piping translation and rotational motion due to static or dynamic loading.

      b) High/moderate-energy boundary such as piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve). The terminal end of such piping is the piping connection to the closed valve.

c) Twelve inch and larger piping connected to the RCL may be modeled with the RCL in the same piping analysis and, therefore, considered a part of the main run. Other branch intersection points are considered a terminal end for the branch line except (1) where the branch and the main piping systems are modeled in the same piping stress analysis and the branch line shown to have a significant effect on the main run behavior (i.e., the nominal size of the branch line is at least one-half of that of the main or the ratio of the moment of inertia of main run pipe to the branch line is less than 10) or (2) where, regardless of size or moment of inertia ratio, the branch lines are short in length and have no significant restraint due to thermal expansion.

2) At intermediate locations where the following conditions are satisfied.

a) The maximum stress range between any two load sets, derived on an elastically calculated basis by Equation (10) and either Equations (12) or (13) of subarticle NB-3653 of ASME Code Section III, under loadings associated with the OBE and normal and upset plant conditions, exceeds 2.4 S_m, or;

b) The cumulative usage factor exceeds 0.1 except for the accumulator safety injection (SI) lines and the pressurizer surge line. For the accumulator safety injection lines and the pressurizer surge line, special analysis was completed and provided to the NRC by References 3.6-15 through 3.6-19. Approval to delete specific postulated breaks for locations with a cumulative usage factor larger than 0.1 was obtained by Reference 3.6-20.

2. ASME Code Section III Class 2 and 3 piping, breaks are postulated to occur at the following locations in each run or branch run:

a. The terminal ends.

b. At all intermediate locations between terminal ends where the primary plus secondary stresses under normal and upset conditions and an OBE event, as calculated on an elastic basis by the sum of Equations (9) and (10) (subarticle NC-3652 of the ASME Code, Section III), exceed 0.8 ($1.2S_H + S_A$). Welded attachments are controlled for high local stresses in accordance with References 3.6-33 through 3.6-35. Therefore, no arbitrary intermediate breaks are postulated.

3. System where a combination of ASME Code Section III Class 1 and Class 2 high-energy piping exists

In cases where both ASME Code Class 1 and Class 2 piping exist between terminal ends, the following apply:

a. If the stress levels and the cumulative usage factor in the ASME Code Class 1 portion and the stress levels in the Class 2 portion exceed the limits specified in items 1 and 2, above, then the breaks are postulated at each of these locations.

4. Non-nuclear high-energy piping

a. Breaks are postulated to occur in non-nuclear piping in the same manner as specified for ASME Code Section III Class 2 and 3 piping if the non-nuclear piping is analyzed and supported to withstand Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) loadings.

b. In the absence of a dynamic seismic analysis, breaks in non-nuclear piping are postulated at the following locations in each run or branch run:

1) Terminal ends

2) Each intermediate fitting (e.g., short- and long-radius elbows, tees and reducers, welded attachments, and valves).

5. Containment penetration piping

a. Main Steam and Feedwater Piping

1) The main steam (MS) and FW System Containment penetration piping including branch connections which are short in length and have no significant restraint to thermal expansion meet the “break-exclusion” requirements of item 5b., below. Figures 3.6.2-7 and 3.6.2-8 show the break exclusion zone for MS and FW system Containment penetration piping, respectively. In addition, mechanistic breaks are postulated in other branches off the MS and FW lines in accordance with Section 3.6.2.1.1.2, above.

2) The isolation valve cubicle housing the break-exclusion portion of MS and FW piping and any safety-related components are designed for a nonmechanistic break occurring anywhere within the break-exclusion zone piping, except in piping and fittings which are associated with the bending and torsional restraints. An assumed single failure of safety related active component concurrent with the nonmechanistic break is not required.

3) The nonmechanistic break is equivalent to one full cross sectional area of undefined type.

4) The penetration structure is capable of withstanding the pressure, temperature, and humidity and flooding transients from the nonmechanistic break.

b. Other Containment penetration piping

Containment penetration piping between the penetration flued head and containment isolation valves, up to and including the restraints that define the terminal ends for the run as stated in item 6), below, may be excluded from postulated breaks (i.e., may be treated as a break-exclusion zone) when all of the following design requirements are met:

1) ASME Code Section III Class 2 Piping: if the following conditions are not met, then requirements listed in Section 3.6.2.1.1.2, above, apply.

   a) The maximum stress ranges as calculated by the sum of Equations (9) and (10) in ASME Section III, subarticle NC-3652, considering operational plant conditions (i.e., sustained loads, occasional loads, and thermal expansion and an OBE event) do not exceed 0.8 (1.2 S_h + S_A).

   b) The maximum stress, as calculated by Equation (9) in subarticle NC-3652 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping, does not exceed $1.8S_h$ except that, following a piping failure outside Containment, the pipe between the isolation valves and the first restraint is permitted higher stresses provided that a plastic hinge is not formed and operability of the valves with such stress is assured in accordance with the requirements of Section 3.9.3.

2) Welded attachments, for pipe supports or other purposes, to these portions of piping are avoided except where detailed stress analyses or tests are performed to demonstrate that the maximum stresses do not exceed the limits defined in item 1), above.

3) The number of circumferential and longitudinal piping welds and branch connections are minimized.

4) The length of these portions of piping is reduced to the minimum length practical.

5) Pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) are not welded directly to the outer surface of the piping (e.g., flued integrally forged pipe fittings may be used) except where all such welds are 100 percent volumetrically examinable as part of the Inservice Inspection (ISI) Program (Section 6.6) and detailed stress analysis is performed to demonstrate that the maximum stresses do not exceed the limits defined in item1), above. Exceptions to the 100 percent volumetric weld examinations (e.g., due to access limitations) are documented in the ISI program.

6) When a break-exclusion zone is established, the terminal end for piping in the zone is consequently extended away from the containment anchor. The terminal end is located adjacent to the restraints that limit the bending and torsion moments exerted on the isolation valve as a consequence of pipe break. These piping restraints are:

a) Located reasonably close to the isolation valves and located to optimize overall piping design.

b) Located, as necessary, to prevent formation of a plastic hinge, following a piping failure, anywhere within the established break exclusion zone.

c) 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.

7) Operability of the isolation valve must be assured for pipe break events where valve operation is required to ensure containment integrity or credit for valve operation is otherwise taken based on the valve integrity and function.

8) Branches originating from the piping run between isolation valves and the containment when analyzed as part of the penetration piping, are subject to the same rules as the main run if treated as part of the no-break region.

9) All piping in the break-exclusion zone must be either of seamless construction with full radiography of all circumferential welds, or of seamed construction with all longitudinal and circumferential welds fully radiographed.

10) All piping greater than 1 in. nominal size in the break exclusion zone shall be subject to an augmented inservice weld examination or as required per the Risk-Informed process for piping outlined in EPRI Topical Report TR-1006937. CN-2894

11) The penetration structure housing a break-exclusion zone portion of high-energy piping and any safety-related components shall be designed for a nonmechanistic break identified in items 5.a.3) and 4), above.

6. A structure that separates a high-energy line outside containment from an essential component is designed to withstand the consequences of the pipe break in the high-energy line which produces the greatest effect at the structure irrespective of the fact that the criteria of Section 3.6.2.1.1 might not require such a break to be postulated.

3.6.2.1.2 ASME Section III and Non-nuclear Piping – Moderate-Energy: Through-wall leakage cracks are postulated in moderate-energy piping including branch runs larger than 1 in. nominal diameter as clarified below:

1. Through-wall leakage cracks are not required to be postulated in those portions of piping between containment isolation valves, provided they meet the requirements of ASME Code, Section III, Subarticle NE-1120, and are designed so that the maximum stress range does not exceed 0.4 (1.2 $S_h + S_A$).

2. Through-wall leakage cracks are not required to be postulated in moderate-energy fluid system piping located in an area where a break in the high-energy fluid system is postulated,

provided that such cracks do not result in environmental conditions more limiting than the high-energy pipe break.

3. Subject to item 4 below, through-wall leakage cracks are required to be postulated in ASME, B&PV Code, Section III Division 1 – Class 2 or 3 piping at locations where the maximum stress range in the piping is greater than 0.4 (1.2 S_h + $S_{A)}$.

4. Individual cracks are not required to be postulated at specific locations determined by stress analyses when a review of the piping layout and plant arrangement drawings shows that the effects of through-wall leakage cracks are isolated or physically remote from structures, systems, and components required for safe shutdown.

5. Through-wall leakage cracks are postulated in nonseismic Category I piping at welded points where the effects might compromise essential equipment or structures.

To simplify analysis, cracks may be postulated to occur everywhere in moderate-energy piping, regardless of the stress analysis results to determine the maximum damage from fluid spraying and flooding, with the consequent hazards or environmental conditions. Flooding effects are determined on the basis of 30-min operator time required to effect corrective actions. Further discussion of internal flooding effects is provided in Section 3.4.3 and 3.4.4.

Cracks in moderate energy ASME Code Class 1 piping are not postulated since there are no ASME Class 1 moderate energy systems. All the ASME Class 1 piping systems are inside the Containment Building and are high energy.

3.6.2.1.3 Types of Breaks/Cracks Postulated:

3.6.2.1.3.1 ASME Section III, class 1 RCL Piping – High-Energy – No breaks are postulated in the ASME Section III, Class 1 primary RCL as discussed in Reference 3.6-14 and paragraph 3.6.2.1.1.1a.

3.6.2.1.3.2 Piping Other than RCL Piping – High-Energy – Breaks are not postulated in the ASME Section III, Class 1 pressurizer surge line and SIS accumulator lines, as discussed in Reference 3.6-21 through 3.6-29, and paragraph 3.6.2.1.1.1.a. For Class 1 piping for which LBB is not applicable, the following types of breaks are postulated to occur at the location determined in accordance with Section 3.6.2.1.1.

1. In piping whose nominal diameter is greater than or equal to 4 in., both circumferential and longitudinal breaks are postulated at each selected break location unless eliminated by comparison of longitudinal and axial stresses with the maximum stress as follows:

   a. If the maximum stress range exceeds the limits specified in Sections 3.6.2.1.1.1.b.2 and 3.6.2.1.1.2.b, but the circumferential stress range is at least 1.5 times the axial stress range, only a longitudinal break is postulated.

   b. If the maximum stress range exceeds the limits specified in Section 3.6.2.1.1.1.b.2 and 3.6.2.1.1.2.b, but the axial stress is at least 1.5 times the circumferential stress range, only a circumferential break is postulated.

c. Longitudinal breaks, however, are not postulated at terminal ends.

2. In piping whose nominal diameter is greater than 1 in. but less than 4 in., only circumferential breaks are postulated at each selected break location.

3. No breaks are postulated for piping whose nominal diameter is 1 in. or less.

3.6.2.1.3.3 Non-Nuclear Piping – High-Energy – the types of breaks postulated for non-nuclear piping are the same as those discussed in Sections 3.6.2.1.3.2. The corresponding break locations are determined in accordance with Section 3.6.2.1.1.4.

3.6.2.1.4 Break/Crack Configuration:

3.6.2.1.4.1 High-Energy Break Configuration – Following a circumferential break, the two ends of the broken pipe are assumed to move clear of each other unless physically limited by piping restraints, structural members, or piping stiffness. The effective cross-sectional (inside diameter flow area of the pipe is used in the jet discharge evaluation. Movement is assumed to be in the direction of the jet reaction initially, with the total path controlled by the piping geometry.

The orientation of a longitudinal break, except when otherwise justified by a detailed stress analysis, is assumed to be oriented (but not concurrently) at two diametrically opposed points on the piping circumference. To maximize the out of plane bending the longitudinal break will be assumed to be perpendicular to the plane of the piping. The flow area of such a break is equal to the cross-sectional flow area of the pipe. Longitudinal and circumferential breaks are not postulated concurrently.

3.6.2.1.4.2 Moderate-Energy Crack Configuration – Moderate-energy crack openings are assumed to be a circular orifice with cross-sectional flow area equal to that of a rectangle one-half the pipe inside diameter in length and one-half pipe wall thickness in width.

3.6.2.2 Analytical Methods to Define Forcing Functions and Response Models - Although the dynamic effects of postulated pipe breaks in the reactor coolant loop primary piping, pressurizer surge line, and SIS accumulator lines can be eliminated for structural design basis (see Section 3.6.2.1.1.1.a), the design verification of certain structures and components may retain the original pipe break loading. For those cases and for breaks for which LBB does not apply, the following subsection describes the methods used in the analysis.

3.6.2.2.1 Forcing Functions for Jet Thrust and Dynamic Model for Piping Response: The fluid conditions at the upstream source and at the break exit dictate the analytical approach and approximations that are used to determine the forcing function. It should be noted that the rise time for the jet thrust is no greater than one millisecond. For most applications, one of the following situations exists:

- Superheated or saturated steam

- Saturated or subcooled water

- Cold water (nonflashing)

Analytical methods for calculation of jet thrust for the above-described situations are discussed in References 3.6-5 and 3.6-6.

For main FW, MS, and reactor coolant surge lines, RELAP 4/5 is used to get the forcing function for the nonlinear time-history pipe jet and whip load analysis. For other lines, Moody's thrust coefficient is used, as specified in Reference 3.6-6.

Nonlinear time-history pipe whip load analysis is a step-by-step determination of piping/whip restraint transient response through time, explicitly including both material (inelastic) and geometric (gap) nonlinear effects. The mathematical models are three-dimensional, lumped-mass models constructed from pipe elements, inelastic energy-absorbing elements, and energy-absorbing device support structure mass and stiffness characteristics. This analysis is performed using Reference 3.6-11, which is based on direct integration of the lumped-mass model's equation of motion.

Dynamic impact and potential rebound effects of the pipe whip problem are explicitly considered in the RELAP 4/5 computer code. Therefore, no additional dynamic amplification factor or rebound effect factor is applied to the non linear time-history results.

The energy balance dynamic analysis method is limited to intermediate-size high-energy lines under 14 inches in diameter. Jet thrust load is taken as the maximum thrust load (with an amplification factor of 1.1) and applied throughout the pipe break event. Maximum restraint device deformation is computed for the energy principle. An appropriate dynamic load factor is then applied to the calculated restraint load for restraint device design.

3.6.2.2.1.1 Time Functions of Jet Thrust Force on RCL Piping – To determine the thrust and reactive force loads to be applied to the RCL during the postulated RCL branch pipe break, it is necessary to have a detailed description of the hydraulic transient. Hydraulic forcing functions are calculated for the RCLs as a result of a postulated RCL branch pipe break. These forces result from the transient flow and pressure histories in the RCS. The calculation is performed in two steps. The first step is to calculate the transient pressure, mass flowrates, 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 RCLs.

The hydraulic model represents the behavior or the coolant fluid within the RCS. 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 the 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 (Ref. 3.6-7) was developed with a capability to provide this information.

The MULTIFLEX computer code calculates the hydraulic transients within the entire primary coolant system. This hydraulic program considers a coupled, fluid-structure interaction by accounting for the deflection of the core support barrel. The depressurization of the system is calculated using the method of characteristics applicable to transient flow of a homogeneous fluid in thermal quilibrium.

The ability to treat multiple flow branches and a large number of mesh points gives the MULTIFLEX code the flexibility required to represent the various flow passages within the primary RCS. The system geometry is represented by a network of one-dimensional flow passages.

The THRUST computer program (Ref. 3.6-8) 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:

$$F = [144A\left(P - 14.7\right) + \left(\frac{m^2}{g\rho\rho\ {}^2{}_m 144}\right)]$$

The symbols and units are as follows:

F = Force, lb_f

A = Aperture area, ft^2

P = System pressure, psia

$\dot{m}$ = Mass flow rate, lbm/sec

ρ = Density, lbm/ft^3

g = Gravitational constant 32.174 ft-lbm/lb-sec^2

$A^2{}_m$ = Mass flow area, ft^2

In the model to compute forcing functions, the RCL system is represented by a model similar to that employed in the blowdown analysis. The entire loop layout is represented in a global coordinate system. Each node is fully described by:

1. Blowdown hydraulic information

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 or 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 apertures in any one node to give a total x force, a total y force, and a total z force. These thrust forces serve as input to the piping/restraint dynamic analysis.

The THRUST code (which uses MULTIFLEX results as input) calculates forces exactly the same way as the (Ref. 3.6-8) STHRUST code, (which uses SATAN [Ref. 3.6-10] results as input).

3.6.2.2.1.2 Dynamic Analysis of the Reactor Coolant Loop Piping and Equipment Supports – The dynamic analysis of the RCL for RCL branch pipe break loadings is described in Section 3.9.

For primary equipment supports, jets from auxiliary lines which impact RCS equipment supports are evaluated to service level D criteria (ASME Subsection NF and Appendix F-1370). Jet loads are added directly to existing faulted condition support loads. For primary loop piping and components, stresses generated from jet loads from auxiliary line jets on Westinghouse-scope piping/equipment, the combination of pressure, deadweight, jet loads are compared against ASME level D condition allowables. Although the dynamic effects of postulated pipe breaks in the reactor coolant loop primary piping, pressurizer surge line, and SIS accumulator lines can be eliminated for structural design basis (see Section 3.6.2.1.1.1.a), the design verification of certain structures and components may retain the original jet impingement loading.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability.- As a result of the application of LBB technology, the dynamic effects of postulated pipe breaks in the reactor coolant loop primary piping, pressurizer surge line, and SIS accumulator lines can be eliminated from the structural design basis (see paragraph 3.6.2.1.1.1.a). However, the design verification of certain components and supports may retain the original pipe break loading. For those cases, the following subsection describes the method used in the analysis.

3.6.2.3.1 Dynamic Analysis Methods to Verify Integrity and Operability for Other than RCL: The analytical methods of Reference 3.6-5, 3.6-6, and 3.6-9 are used to determine the jet impingement effects and loading effects applicable to components and systems resulting from postulated pipe breaks and cracks. Note that for short periods of time, the pressure and enthalpy in certain systems will be higher than full or normal power operation (i.e., 102 percent power). However, the full power mode establishes the maximum demands of safety systems in the event of a postulated pipe rupture. Other modes of normal operation have reduced needs for safety systems to bring the plant to a safe shutdown. Therefore, the full power operation mode is used to determine the thermodynamics state in the piping system for the calculation of fluid reaction forces.

3.6.2.3.2 Dynamic Analysis Methods to Verify Integrity and Operability for the RCL:

3.6.2.3.2.1 General – A LOCA is assumed to occur for a branch line break down to the second normally open automatic isolation valve (Case II, Figure 3.6.2-1) on outgoing lines and down to and including the second check valve (Case III, Figure 3.6.2-1) on incoming lines normally with flow. A pipe break beyond the second check valve does not result in an uncontrolled loss of reactor coolant if either of the two valves in the line closes.

Periodic testing of the valves capability 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, Figure 3.6.2-1), a LOCA is assumed to occur for pipe breaks on the reactor side of the valve.

Branch lines connected to the RCL are defined as large strictly for the purpose of pipe break criteria when they have an inside diameter greater than 4 in. up to the largest connecting line. Rupture of these lines results in a rapid blowdown from the RCL, and protection is basically provided by the accumulators and the low-head safety injection (LHSI) pumps.

Branch lines connected to the RCL are defined as small for the purpose of pipe break analysis if they have an inside diameter equal to or less than 4 in. This size is such that Emergency Core Cooling System (ECCS) analyses, using realistic assumptions, show that no fuel cladding damage is expected for a break area of up to 12.5 in.^{2} corresponding to 4in. inside diameter piping.

Engineered safety features (ESFs) are provided for core cooling and boration, pressure reduction, and activity confinement in the event of a LOCA or steam or FW line break accident to ensure that the public is protected in accordance with 10CFR100 Guidelines. The original design basis postulated pipe break locations in the RCL are described in Reference 3.6-1. A detailed fracture mechanics evaluation, as described in Reference 3.6-14, demonstrates that the probability of rupturing the RCL piping is extremely low under design basis conditions. Therefore, postulated RCL ruptures and the associated dynamic effects are not included in the design basis. However, to retain high safety margins, these safety systems are designed to provide protection for an RCS pipe rupture of a size up to and including a double-ended severence of the RCS main loop.

To assure the continued integrity of the essential 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 leaktightness is not decreased below the design value if the break leads to a LOCA (1)[1].

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

    a. A pipe break which is not a LOCA or steam/FW line break will not cause a LOCA or steam/FW line break.

    b. An RCL branch pipe break will not cause a steam or feedwater system pipe break, and vice versa, in excess of small lines which are not required to function following accidents.

    Exceptions to these criteria may be made if specific evaluations show no adverse effects occur to accident mitigation and recovery systems.

3.6.2.3.2.2 Large RCL Branch Piping – Large branch line piping, as defined in Section 3.6.2.3.2.1, is restrained to meet the following criteria in addition to items 1 through 3 of Section 3.6.2.3.2.1 for a pipe break resulting in a LOCA:

[1] The Containment is here defined as the Containment structure liner and penetrations and the steam generator shell, the steam generator steam side instrumentation connections, the steam, FW, blowdown, and steam generator drain pipes within the Containment structure.

1. Propagation of the break to the unaffected loops is limited to small instrument or sample lines to ensure 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. The criterion is voluntarily applied so as not to substantially increase the severity of the LOCA.

   Exceptions to these criteria may be made if specific evaluations show no adverse effects occur to accident mitigation and recovery systems.

3.6.2.3.2.3 Small RCL Branch Lines – Should one of the small pressurized RCL branch lines, as defined in Section 3.6.2.3.2.1, fail and result in a LOCA, the piping is restrained or arranged to meet the following criteria in addition to items 1 through 3 of Section 3.6.2.3.2.1:

1. Break propagation is limited to small instrument or sample lines in the unaffected leg and loops; i.e., propagation to the other leg of the affected loop and to the other loops is minimized. Damage to the high-head safety injection (HHSI) lines connected to the other leg of the affected loop or to the other loops is prevented.

2. Propagation of the break in the affected leg is permitted but must be limited to a total break area of 12.5 in2.

   Exceptions to these criteria may be made if specific evaluations show no adverse effects occur to accident mitigation and recovery systems.

3.6.2.3.2.4 Design and Verification of Adequacy of RCL Components and Supports – The original design basis postulated pipe break locations in the RCL are described in Reference 3.6-1. The primary RCL components and supports design were based on these postulated break locations. A detailed fracture mechanics evaluation, as described in References 3.6-14, and 3.6-21 through 3.6-29, demonstrates that the probability of rupturing the reactor coolant loop primary piping, pressurizer surge line and SIS accumulator lines is extremely low under design basis conditions. Therefore, postulated ruptures in the RCL, surge line, and SIS accumulator lines, and the following associated dynamic effects are not included in the design basis: Missile generation, pipe whip, break reaction forces, jet impingement forces, decompression waves within the ruptured pipe, and pressurization in cavities, subcompartments and compartments. The dynamic effects from ruptures in Class 1 branch lines not covered by LBB and other high energy piping are reviewed to verify that the effects are bounded by the current analyses.

3.6.2.3.3 Type of Pipe Whip Restraints: As discussed in paragraph 3.6.2.1.1.1.a, the dynamic effects of postulated pipe breaks in the reactor coolant loop primary piping, pressurizer surge line, and the three SIS accumulator lines can be eliminated from the structural design basis. Therefore, whip restraints for these piping systems are not required.

3.6.2.3.3.1 Pipe Whip Restraints – To satisfy varying requirements of available space, permissible pipe deflection, and equipment operability, the restraints are designed as a combination of an energy-absorbing element and a restraint structure suitable for the geometry required to pass the restraint load from the whipping pipe to the main building structure.

The restraint structure is typically a structural steel frame or truss and the energy-absorbing element is usually either stainless steel U-bars or energy-absorbing material as described below:

1. Stainless Steel U-Bar

   This type consists of one or more U-shaped, upset-threaded rods of stainless steel looped around the pipe but not in contact with the pipe to allow unimpeded pipe motion during seismic and thermal movement of the pipe. At rupture, the pipe moves against the U-bars, which absorb the kinetic energy of the pipe motion by yielding plastically. A typical example of a U-bar restraint is shown in Figure 3.6.2-3.

2. Energy Absorbing Material

   This type of restraint consists of a crushable, stainless steel, internally honeycomb-shaped element designed to yield plastically under impact of the whipping pipe. A design hot position gap is provided between the pipe and the energy-absorbing material to allow unimpeded pipe motion during seismic and thermal pipe movements. A typical example of an energy-absorbing material restraint is shown in Figure 3.6.2-4.

3. Five-Way Restraint

   A five-way restraint is utilized to protect the main steam isolation valves (MSIVs) and main FW isolation valves in the event of postulated pipe rupture outside the Containment. This restraint is designed so that postulated pipe breaks beyond the five-way restraint will not result in stresses greater than 1.8 S_h being transmitted to the piping between the isolation valve and containment penetration or formation of a plastic hinge between the isolation valve and the restraint.

3.6.2.3.3.2 Restraints for RCL – As discussed in Reference 3.6-14 and Section 3.6.2.1.1.1a, RCL ruptures and the associated dynamic effects are not included in the design bases. RCL pipe restraints are no longer required.

3.6.2.3.4 Analytical Methods:

3.6.2.3.4.1 Pipe Whip Restraints –

1. Location of Restrains

   a. For purposes of determining pipe hinge length and thus locating the pipe whip restraints, the plastic moment of the pipe may be determined in the following manner:

   $$M_p = 1.1\, z_p S_y$$

   where:

   z_p = Plastic section modulus of pipe = $\frac{4}{3}(r_o^3 - r_i^3)$

r_i = inside radius of pipe

r_o = outside radius of pipe

S_y = Yield stress at pipe operating temperature

1.1 = 10-percent factor to account for strain hardening (for T < 400°F)

Alternatively, the load carrying capacity of the pipe may be determined by a suitable analytical model per Reference 3.6-9.

Pipe whip restraints are located as close to the axis of the reaction thrust force break as practicable. Pipe whip restraints are generally located so that a plastic hinge does not form in the pipe. If, due to physical limitations, pipe whip restraints are located so that a plastic hinge can form, the consequences of the whipping pipe and the jet impingement effect are further investigated. Lateral guides are provided where necessary to predict and control pipe motion.

b. Generally, restraint are designed and located with sufficient clearances between the pipe and the restraint such that they do not interact and cause additional piping stresses. A design hot position gap is provided that will allow maximum predicted thermal, seismic, and seismic anchor movement displacements to occur without interaction.

Exception to this general criterion may occur when a pipe support and restraint are incorporated into the same structural steel frame, or when a zero design gap is required. In these cases the restraint is included in the piping analysis.

c. In general, the restraints do not prevent the access required to conduct ISI of piping welds. When the location of the restraint makes the piping welds inaccessible for inservice inspection, a portion of the restraint is made removable to provide accessibility.

2. Analysis and Design

Analysis and design of pipe whip restraints for postulated pipe break effects are in accordance with Reference 3.6-5. Specifically, the following criteria are adopted in analysis and design:

a. Pipe whip restraints are designed based on energy absorption principles by considering the elastic-plastic, strain-hardening behavior of the materials used.

b. A rebound factor of 1.1 is applied to the jet thrust force (when static analyses are performed).

c. Except in cases where calculations are performed to verify that a plastic hinge is formed, the energy absorbed by the ruptured pipe is conservatively assumed to be zero; i.e., the thrust force developed goes directly into moving the broken pipe and is not reduced by the force required to bend the pipe.

d. In elastic-plastic design, limits for strains are as follows:

ε = Allowable strain used in design.

1) Stainless Steel U-Bars

$\varepsilon = 0.5\ \varepsilon_{\mu}$

where:

ε_{μ} = ultimate uniform strain of stainless steel (strain at ultimate stress).

2) Energy-Absorbing Material

$\varepsilon = 0.8\ \varepsilon_{\mu}$

where:

ε_{μ} = maximum strain at uniform crushable strength.

e. A dynamic increase factor is used for steel which is designed to remain elastic.

3.6.2.3.4.2 RCL Restraints – As discussed in Reference 3.6-14 and Section 3.6.2.1.1.1a, RCL ruptures and the associated dynamic effects are not included in the design bases. RCL pipe restraints are no longer required.

3.6.2.4 Protective Assembly Design Criteria.

3.6.2.4.1 Jet Impingement Barriers and Shields: Barriers and shields, which may be of either steel or concrete construction, are provided to protect essential equipment, including instrumentation, from the effects of jet impingement resulting from postulated pipe breaks. Barriers differ from shields in that they may also accept the impact of whipping pipes. Barriers and shield include walls, floors, and structures specifically designed to provide protection from postulated pipe breaks. Barrier and shield design is based on the methods of Reference 3.6-5, Section 3.0, and the elastic-plastic methods for dynamic analysis included in Reference 3.6-12. Design criteria and loading combinations are in accordance with Sections 3.8.3 and 3.8.4.

3.6.2.4.2 Auxiliary Guardpipes: The use of guardpipes has been minimized by plant arrangement and routing of high-energy piping. Where they are used, guardpipes are designed to withstand all dynamic and environmental effects of postulated breaks of the enclosed pipe. Auxiliary guardpipes are used only if inservice inspection requirements can be satisfied. Design criteria, loading combinations, and methods of analysis are similar to those for barriers and shields described in Section 3.6.2.4.1.

3.6.2.5 Material Submitted for the Operating License Review.

3.6.2.5.1 Piping Systems Other than RCL: Pipe break locations are obtained in accordance with the criteria of Section 3.6.2.1. As discussed in paragraph 3.6.2.1.1.1.a, the dynamic effects of postulated pipe rupture have been eliminated from structural design basis for the pressurizer surge line and the three SIS accumulator lines.

Figure 3.6.1-1 identifies the break locations in high-energy piping. The stress results utilized to determine the break types and locations are given in Table 3.6.2-1. Associated stress nodes are shown in Figure 3.6.1-1. High-energy pipe break effects analysis for a selected portion of the plant are discussed in Appendix 3.6.B. Appendix 3.6.B also references the appropriate sheet of applicable high-energy lines shown in Figure 3.6.1-1.

Moderate-energy piping crack locations are defined in Section 3.6.2.1.2. Evaluation of the flooding effects of moderate-energy cracks is discussed in Section 3.4.3 and 3.4.4.

The augmented ISI plan is discussed in Section 6.6.

Pipe whip restraints are designed in accordance with Section 3.6.2.3. Pipe whip restrain location and orientation for each high-energy break are shown in Figure 3.6.1-1. Barriers and shields are designed in accordance with the criteria of Section 3.6.2.4. Jet thrust and impingement forces were determined in accordance with Reference 3.6-5. Reaction forces for each pipe whip restraint are presented in Figure 3.6.1-1.

3.6.2.5.2 Reactor Coolant Loop:

1. The original design basis postulated pipe break locations in the RCL are described in Reference 3.6-1. A detailed fracture mechanics evaluation, as described in Reference 3.6-14, demonstrates that the probability of rupturing the RCL piping is extremely low under design basis conditions. Therefore, postulated RCL ruptures and the associated dynamic effects are not included in the design basis.

2. RCL pipe whip restraints are not required.

3. Design loading combinations and applicable criteria for ASME Class 1 components and supports are provided in Section 3.9. Pipe rupture loads include not only the jet thrust forces acting on the piping but also jet impingement loads on the primary equipment supports. Although the dynamic effects of postulated pipe breaks in the reactor coolant loop primary piping, pressurizer surge line, and SIS accumulator lines can be eliminated for structural design basis (see Section 3.6.2.1.1.1.a), the design verification of certain structures and components may retain the original pipe break loading.

REFERENCE

Section 3.6:

3.6-1 "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop", WCAP-8082-P-A, (Proprietary) and WCAP-8172-A (Non-Proprietary), January 1975.

3.6-2 "Subcompartment Pressure Analyses", BN-TOP-4, Rev. 1, Bechtel Power Corporation, October 1977.

3.6-3 USNRC BTP MEB 3-1 Postulated Break and Leakage Locations in Fluid System Piping Outside Containment. Branch Technical Position attached to SRP 3.6.2, November 24, 1975.

3.6-4 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, 1974 and 1975 Winter Addenda and other Addenda as appropriate.

3.6-5 "Design for Pipe Break Effects", Bechtel Power Corporation, BN-TOP-2, Revision 2, May 1974.

3.6-6 Moody, F. J., "Fluid Reaction and Impingement Loads". Paper presented at the ASCE Specialty Conference, Chicago, December 1973.

3.6-7 "MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics", WCAP-8708 (Proprietary), February 1976, and WCAP-8709 (Non-Proprietary), February 1976.

3.6-8 "Documentation of Selected Westinghouse Structural Analysis Computer Codes", WCAP-8252, Revision 1, May 1977.

3.6-9 ANSI/ANS – 58.2, "American National Standard Design Basis for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Rupture", December 1980.

3.6-10 Borderlon, F.M., "A Comprehensive Space-Time Dependent Analysis of Loss of Coolant (SATAN IV Digital Code)" WCAP-7263, Proprietary (August 1971) and WCAP-7750, Non-Proprietary (August 1971).

3.6-11 "PIPERUP" – Pipe Rupture Analysis Program, ME351, June 24, 1982.

3.6-12 Biggs, J.M., Introduction to Structural Dynamics, McGraw-Hill Book Company, New York 1964.

3.6-13 NUREG/CR 2913, "Two Phase Jet Loads", dated January 1983.

REFERENCE (Continued)

Section 3.6:

3.6-14 "Technical Bases for Eliminating Large Primary Loop Pipe Ruptures as the Structural Design Basis for the South Texas Project", WCAP-10559, Proprietary (May 1984) and WCAP-10560, Non-Proprietary (May 1984).

3.6-15 Houston Lighting & Power Company Letter to NRC, M. R. Wisenburg to H. L. Thompson, Jr., February 28, 1986, ST-HL-AE-1611.

3.6-16 Houston Lighting & Power Company Letter to NRC M. R. Wisenburg to V. S. Noonan, September 17, 1986, ST-HL-AE-1758.

3.6-17 "Cumulative Usage Factor Criterion for Break Postulation for South Texas Units 1 and 2", WCAP-11327, Proprietary (October, 1986) and WCAP-11328, Nonproprietary (October, 1986); submitted by Houston Lighting & Power Company Letter to NRC, M. R. Wisenburg to V. S. Noonan, October 31, 1986 ST-HL-AE-1793.

3.6-18 Houston Lighting & Power Company Letter to NRC, M. R. Wisenburg to V. S. Noonan, December 8, 1986, ST-HL-AE-1830.

3.6-19 Houston Lighting & Power Company Letter to NRC, M. R. Wisenburg to V. S. Noonan, December 15, 1986, ST-HL-AE-1845.

3.6-20 USNRC Letter to Houston Lighting & Power Company, V. S. Noonan to J. H. Goldberg, December 31, 1986, ST-AE-HL-91085.

3.6-21 "Technical Bases for Eliminating Pressurizer Surge Line Ruptures as the Structural Design Basis for South Texas Project," WCAP-10489, February 1984.

3.6-22 "Additional Information in Support of the Elimination of Postulated Pipe Ruptures in the Pressurizer Surge Lines of South Texas Project Units 1 and 2," WCAP-11256, September 1986.

3.6-23 "Technical Bases for Eliminating Pressurizer Surge Line Ruptures as the Structural Design Basis for South Texas Project Units 1 and 2," WCAP-11256, Supplement 1, November 1986.

3.6-24 Houston Lighting & Power Company letter to the NRC, dated April 28, 1987, with Attachment 1 entitled "Additional Information in Support of the Elimination of Postulated Pipe Ruptures in the Pressurizer Surge Lines of South Texas Project Units 1 and 2."

REFERENCE (Continued)

Section 3.6:

3.6-25 Technical Bases for Eliminating Class 1 Accumulator Line Rupture as the Structural Design Basis for South Texas Project Units 1 and 2," WCAP 11383, January 1987.

3.6-26 Technical Bases for Eliminating Accumulator Class 1 Low Pressure Line Rupture as the Structural Design Basis for South Texas Project Units 1 and 2," WCAP 11351, March 1987.

3.6-27 Houston Lighting & Power Company letter to the NRC, dated May 18, 1987, with Attachment 1 entitled "Additional Information in Support of the Elimination of Postulated Pipe Ruptures in the Accumulator Lines of South Texas Project Units 1 and 2."

3.6-28 Houston Lighting & Power Company letter to the NRC, dated May 29, 1987, with Attachment 1 entitled "Additional Information in Support of the Elimination of Postulated Pipe Ruptures in the Accumulator Lines of South Texas Project Units 1 and 2 (10-inch and 8-inch RHR Piping Attached to the Accumulator Line)"

3.6-29 "Technical Bases for Eliminating Rupture of Accumulator Line and its Attached RHR Piping from the Structural Design Basis for South Texas Projects Units 1 and 2," WCAP 11555, September 1987.

3.6-30 "Safety Evaluation Report Related to the Operation of South Texas Project Units 1 and 2," NUREG-0781, Supplement No. 2, Docket Nos. 50-498 and 50-499, January 1987.

3.6-31 "Safety Evaluation Report Related to the Operation of South Texas Project Units 1 and 2," NUREG-0781, Supplement No. 4, Docket Nos. 50-498 and 50-499, July 1987.

3.6-32 "Safety Evaluation Report Related to the Operation of South Texas Project Units 1 and 2," NUREG-0781, Supplement No. 5, Docket Nos. 50-498 and 50-499, March 1988.

3.6-33 Houston Lighting & Power Company letter to NRC, M. R. Wisenburg to Attention: Document Control Desk, April 8, 1987, ST-HL-AE-2025.

3.6-34 Houston Lighting & Power Company letter to NRC, M. R. Wisenburg to Attention: Document Control Desk, July 8, 1987, ST-HL-AE-2290.

3.6-35 Houston Lighting & Power Company letter to NRC, M. A. McBurnett to Attention: Document Control Desk, February 17, 1989, ST-HL-AE-2979.

REFERENCE (Continued)

Section 3.6:

3.6-36 "Evaluation of Thermal Stratification for the South Texas Units 1 and 2 Pressurizer Surge Line," WCAP-12067, Revision 1 (Proprietary), and WCAP-12087 (Non-Proprietary), January 1989; and WCAP-12607, Revision 1, Supplement 1 (Proprietary), February 1989.

3.6-37 "Safety Evaluation Report Related to the Operation of South Texas Project, Unit 2," NUREG-0781 Supplement No. 7, Docket No. 50-499, March 1989

TABLE 3.6.1-1

ESSENTIAL, HIGH ENERGY, AND MODERATE – ENERGY SYSTEMS

System	Essential[a] Systems	High[b] Energy	Moderate[c] Energy
Reactor Coolant System	X	X	
Main Steam System	X	X	
Main Feedwater System	X	X	
Auxiliary Feedwater System	X	X	X
Steam Generator Blowdown System	X	X	X
Auxiliary Steam System		X	
Chemical and Volume Control System	X	X	X
Residual Heat Removal System	X	X	X
Safety Injection System	X	X	X
Extraction Steam System		X	
Heater Drips System		X	
Turbine Bypass System		X	
Turbine Gland Sealing System		X	
Compressed Air System for Diesel			X
Generator Starting System			X
Containment Systems including:			
Containment Vessel	X		
Containment Penetrations	X		
Containment Isolation Valves	X		
Containment Sump	X		
Reactor Containment Fan Coolers	X		X
Containment Purge System	X		X
Auxiliary Cooling Water System			X
Circulating Water System			X
Main Condenser Evacuation System			X
Fire Protection Systems			X
Demineralized Water Makeup System			X
Potable and Sanitary Water System			X
Condensate Storage System			X
Diesel–Generator Closed Cooling			X
Water System	X		X
Diesel-Generator Lubricating Oil			
System	X		X

TABLE 3.6.1-1 (Continued)

ESSENTIAL, HIGH ENERGY, AND MODERATE – ENERGY SYSTEMS

System	Essential[a] Systems	High[b] Energy	Moderate[c] Energy
Diesel Fuel Oil Storage and Transfer System			X
Turbine Lube Oil System			X
Stator Liquid Cooling System			X
Hydrogen Seal Oil System			X
Boron Recycle System			X
Containment Spray System	X		X
Essential Cooling Water Systems	X		X
Component Cooling Water System	X		X
Spent Fuel Pool Cooling and Cleanup System	X		X
Reactor Makeup Water System			X
Liquid Radwaste System			X
Chilled Water System			X
Post-Accident Monitoring System	X	X	X
Radiation Monitoring	X	X	X
Reactor Vessel Head Vent System	X	X	

a. Not all essential systems are required for postulated piping failures; e.g., the containment spray system is essential for loss-of-coolant accident and main steam line break inside containment, but is nonessential for piping failure outside containment. Not all portions of essential systems are required for postulated piping failure; e.g., the main steam system is only essential from the steam generator to the main steam isolation valves, including the safety and atmospheric steam relief valves.
b. Not all portions of high-energy systems contain high-energy fluid.
c. During the initial phase of cooldown, the residual heat removal system is a high-energy system. For interaction with the redundant train, the residual heat removal system is considered a dual-purpose, moderate-energy system (Section 3.6.1.1[7]).

TABLE 3.6.1-2

DESIGN COMPARISON TO POSITIONS OF NRC BRANCH TECHNICAL POSITION ASB 3-1

Branch Technical Position ASB 3-1	STPEGS Design
B.1 Plant arrangement	B.1 Conforms. See Section 3.6.1.3
Protection of essential systems and components against postulated piping failures in high or moderate-energy fluid systems that operate during normal plant conditions and that are located outside of containment should be provided.	
	B.1.a:Conform. See Section 3.6.1.3.2.(1)
	B.1.a.(1) Partial Conformance as follows: The essential equipment located in the main steam and main feedwater penetration areas is designed to be protected from or qualified to the environmental effects (compartment pressure, temperature, humidity, and flooding) resulting from a full circumferential break (single area) in the main steam or main feedwater lines. The essential equipment is designed to be protected from the jet impingement and pipe whip effects resulting from a full circumferential break postulated in branch lines associated with the main steam or main feed-water lines.
	B.1.a.(2) Conforms. See Section 3.6.1.2

TABLE 3.6.1-2 (Continued)

DESIGN COMPARISON TO POSITIONS OF NRC BRANCH TECHNICAL POSITION ASB 3-1

Branch Technical Position ASB 3-1		STPEGS Design	
		B.1.b	Conforms. See Section 3.6.1.3.2.(2)
		B.1.c	Conforms. See Sections 3.6.1.3.2.(2); 3.6.1.3.2.(3);and 3.6.2.3.
		B.1.c.1(a)	Conforms. As part of the design process, the restraint gap is verified large enough to accommodate thermal, seismic, and seismic anchor movements and other occasional loads.
		B.1.c.1(b)	Partial conformance. See Section 3.6.2.3.3.1. Additionally, final pipe whip restraint gap will be verified during hot-functional testing and thus will account for any differential settlement. Pipe relaxation is not specifically considered in the STPEGS design.
		B.1.c.1(c)	See response to items (a) and (b) above.
		B.1.c.(2)	Partial conformance. Restraints which do not have adequate inservice inspection pipe weld space requirements are made removable or exceptions are documented in the ISI program. See Section 3.6.2.1.1(5.b.5).
B.2	Design Features.		
B.2.a	Essential systems and components should be designed to meet the seismic design requirements of Regulatory Guide 1.29.	B.2.a	Conforms, as described in Section 3.2.

TABLE 3.6.1-2 (Continued)

DESIGN COMPARISON TO POSITIONS OF NRC BRANCH TECHNICAL POSITION ASB 3-1

Branch Technical Position ASB 3-1		STPEGS Design	
B.2.b	Protective structures or compartments, fluid and other protective measures.	B.2.b	Conforms. See Sections 3.8.3 and 3.8.4 for loading combinations. See Sections 3.6.1.1 and 3.6.1.3 for piping restraints and protection measures.
B.2.c	Fluid system piping in containment penetration areas should be designed to meet the break exclusion provisions contained in item B.1.b of BTP MEB 3-1.	B.2.c	Conforms. High-energy piping is designed as per B.1.b or MEB 3-1. Moderate-energy piping is designed as per B2.B of MEB 3-1. For further information, refer to B.1.a.(1) above and Sections 3.6.2.1.1.(5) and 3.6.2.1.2.
B.2.d	Piping classification as required by NRC Regulatory Guide 1.26 should be maintained.	B.2.d	Conforms. See Section 3.6.2.1.1.
B.3	Analyses and Effects of Postulated Piping Failures.	B.3.a	Conforms. See Section 3.6.1.1.(5), 3.6.1.2; and 3.6.1.3.
	B.3.b.(1)	Conforms. See Section 3.6.1.1.(5).	
	B.3.b.(2)	Conforms. See Section 3.6.1.1.(6).	
	B.3.b.(3)	Conforms. See Section 3.6.1.1.(7).	
	B.3.b (4)	Conforms. See Section 3.6.1.1.(8).	
	B.3.c	Conforms. See Section 3.6.1.2 and 3.6.1.3.3.(6.b).	
	B.3.d	Conforms. See Section 3.6.1.1.(8.b).	

TABLE 3.6.1-3

DESIGN COMAPRISON TO NRC BRANCH TECHNICAL POSITION MEB 3-1 1

Branch Technical Position MEB 3-1		STPEGS Design	
B.1	High-Energy Fluid System Piping		
B.1.a	Fluid systems separated from essential systems and components.	B.1.a	Conforms. See Section 3.6.1.3.2.1.
B.1.b	Fluid system piping in containment penetration areas.	B.1.b	Partial conformance. See Section 3.6.2.1.1.5.
B.1.b.(1)(a)-(c)			There is no Class 1 piping in Containment penetration areas in the STPEGS.
B.1.b(1)(d)			Conforms. See Section 3.6.2.1.1.5.
B.1.b.(1)(e)			Conforms. For further discussion see Section 3.6.2.1.1.5.
B.1.b.(2)			Conforms. See Section 3.6.2.1.1.5.
B.1.b.(3)			Conforms. See Section 3.6.2.1.1.5.
B.1.b.(4)			See conformance statement to ASB 3-1 position B.2.c.(1) and Section 3.6.2.1.1(5.b.4).
B.1.b.(5)			High-energy containment flued head penetrations are integrally forged piped fittings. Pipe whip restraints do not require welding directly to the outer surface of the piping, except where such welds are 100-percent volumetrically examined in service and a review for local stresses is performed.
B.1.b.(6)			No guard pipes in high energy lines.

TABLE 3.6.1-3 (Continued)

DESIGN COMAPRISON TO NRC BRANCH TECHNICAL POSITION MEB 3-1

Branch Technical Position MEB 3-1		STPEGS Design	
B.1.b.(7)			Conforms. See Section 3.6.2.1.1.5.b.5 and 10.
B.1.c	Postulation of pipe rupture in areas other than containment penetration.	B.1.c.	Conforms. See Section 3.6.2.1.1.
		B.1.c.(1)(a)-(d)	Partial Conformance. Break locations are limited to the stress determined breaks and terminal en breaks. See Section 3.6.2.1.1.1.
		B.1.c.(2)	Partial Conformance. Break locations are limited to the stress determined breaks and terminal end breaks. See Section 3.6.2.1.1.2.
		B.1.c.(3)	Conforms. See Section 3.6.2.1.1.4.
		B.1.c.(4)	Partial Conformance. Conformance to structures separating a high energy line from an essential component is limited to high energy lines outside containment. However, structures inside containment are designed for the dynamic effects of postulated mechanistic breaks. See Sections 3.6.2.1.1(1) and 3.6.2.1.1(6).
		B.1.d.	Conforms. See Section 3.6.2.5.
		B.1.e.	Partial Conformance. In lieu of postulating high energy leakage cracks for environmental effects, certain non-mechanistic full circumferential breaks are postulated to establish the environmental conditions inside Containment. The bulk containment effects due to leakage cracks are enveloped by these breaks.

TABLE 3.6.1-3 (Continued)

DESIGN COMAPRISON TO NRC BRANCH TECHNICAL POSITION MEB 3-1

Branch Technical Position MEB 3-1			STPEGS Design
B.2	Moderate-Energy Fluid System Piping	B.2.a.	Conforms. See Section 3.6.2.1.2.
		B.2.b.	Conforms. See Section 3.6.2.1.2.
		B.2.c.(1)-(2)	Conforms. See Section 3.6.2.1.2.
		B.2.d.	Conforms. See Section 3.6.2.1.2.
		B.2.e.	Conforms. See Section 3.6.1.1.1
B.3	Type of Breaks and Leakage Cracks in Fluid System Piping.	B.3.a.(1)	Conforms. See Section 3.6.2.1.3.
		B.3.a.(2)	Conforms. See Section 3.6.2.1.3.
		B.3.a.(3)	Conforms. See Section 3.6.2.1.4.1
		B.3.a.(4)	See Section 3.6.2.2. 1
		B.3.a.(5)	Conforms. See Section 3.6.1.1.10
		B.3.b.	Conforms. See Section 3.6.2.1.3.
		B.3.c.	Conforms. See Section 3.6.2.1, 3.6.1.3.3.6b, and 3.6.1.2.

TABLE 3.6.2-1

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS
(FOR UNIT 1 ONLY)

System: Main Steam System
(Inside Containment)

Problem No.: MS-01
Revision: 2

UFSAR Figure: Figure 3.6.1-1 (Sheet 1A)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
1A	18,096	0.479	SG Nozzle	TE/C
60	27,641	0.731	Containment Penetration (M-2)	TE/C[3]

1. Ratio = Total stress/37,800 psi
2. TE = Terminal End
   C = Circumferential
   L = Longitudinal
3. Highest relative stress point

TABLE 3.6.2-1

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Main Steam System
(Inside Containment)

Problem No.: MS-01
Revision: 2

UFSAR Figure: Figure 3.6.1-1 (Sheet 1A)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
1A	19,206	0.508	SG Nozzle	TE/C
60	22,441	0.594	Containment Penetration (M-2)	TE/C[3]

1. Ratio = Total stress/37,800 psi
2. TE = Terminal End
   C = Circumferential
   L = Longitudinal
3. Highest relative stress point

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Main Steam System
(Inside Containment)

Problem No.: MS-02
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 1B)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
1	17,492	0.475	SG Nozzle	TE/C
60	18,864	0.499	Containment Penetration (M-3)	TE/C
C02M	30,043	0.795	Elbow	(3)

1. Ratio = Total stress/37,800 psi
2. TE = Terminal End
   C = Circumferential
   L = Longitudinal
3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Main Steam System (Inside Containment)

Problem No.: MS-03
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 1C)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
1C	23,730	0.628	SG Nozzle	TE/C
M4	20,850	0.552	Containment Penetration (M-4)	TE/C
B2M	27,924	0.739	Elbow	(3)

1. Ratio = Total stress/37,800 psi
2. TE = Terminal End
   C = Circumferential
   L = Longitudinal
3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Main Steam System
(Inside Containment)

Problem No.: MS-04
Revision: 2

UFSAR Figure: Figure 3.6.1-1 (Sheet 1D)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2)
1D	20,250	0.536	SG Nozzle	TE/C
M1	17,682	0.468	Containment Penetration (M-1)	TE/C
B2M	32,769	0.867	Elbow	(3)

1. Ratio = Total stress/37,800 psi
2. TE = Terminal End
   C = Circumferential
   L = Longitudinal
3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Main Feedwater System
(Inside Containment)

Problem No.: FW-01
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 2E)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2][4]
N02	40,554	1.028	SG Nozzle	TE/C
005M	37,985	0.962	Elbow	(3)
120	20,784	0.641	Containment Penetration (M-6)	TE/C

1. Ratio = Total stress/stress limit $= \dfrac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \dfrac{\text{Total stress}}{32{,}400\ \text{psi}}$ for Node 120;

   $= \dfrac{\text{Total stress}}{39{,}468\ \text{psi}}$ for Nodes N02 & 005M

2. TE = Terminal End   C = Circumferential

   IM = Intermediate   L = Longitudinal

2. Highest relative stress point other than terminal ends: No break postulated

4. Arbitrary intermediate breaks are not postulated.

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Main Feedwater System
(Inside Containment)

Problem No.: FW-01
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 2A)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2], [4]
N02	39,291	0.996	SG Nozzle	TE/C
005M	36,588	0.927	Elbow	(3)
120	21,599	0.667	Containment Penetration (M-6)	TE/C

1. Ratio = Total stress/stress limit $= \frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$ for Node 120;

   $= \frac{\text{Total stress}}{39{,}468\ \text{psi}}$ for Nodes N02 & 005M

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated
4. Arbitrary intermediate breaks are not postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Main Feedwater System
(Inside Containment)

Problem No.: FW-02
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 2F)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2][4]
N02	29,935	0.616	SG Nozzle	TE/C
004	30,328	0.768	Elbow	(3)
110	17,870	0.552	Containment Penetration (M-7)	TE/C

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8\,(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$ for Node 110;

   $= \frac{\text{Total stress}}{48{,}600\ \text{psi}}$ for Node N02;

   $= \frac{\text{Total stress}}{39{,}468\ \text{psi}}$ for Node 004

2. TE = Terminal End   C = Circumferential

   IM = Intermediate   L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated

4. Arbitrary intermediate breaks are not postulated.

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Main Feedwater System
(Inside Containment)

Problem No.: FW-02
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 2B)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2), (4)
N02	29,376	0.604	SG Nozzle	TE/C
004	30,110	0.763	Elbow	(3)
110	17,200	0.531	Containment Penetration (M-7)	TE/C

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$ for Node 110;

$$= \frac{\text{Total stress}}{48{,}600} \text{ for N02;}$$

$$= \frac{\text{Total stress}}{39{,}468} \text{ for 004}$$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated

4. Arbitrary intermediate breaks are not postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Main Feedwater System
(Inside Containment)

Problem No.: FW-03
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 2G)

Node	Total Stress (psi)	Ratio$^{(1)}$	Location	Type of Postulated Breaks$^{(2)(4)}$
N02	17,326	0.357	SG Nozzle	TE/C
005M	24,163	0.612	Elbow	(3)
110	16,783	0.518	Containment Penetration (M-8)	TE/C

1. Ratio = Total stress/stress limit = $\dfrac{\text{Total stress}}{.8(1.2\,S_h + S_A)} = \dfrac{\text{Total stress}}{32{,}400 \text{ psi}}$ for Node 110;

   $= \dfrac{\text{Total stress}}{48{,}600 \text{ psi}}$ for Node N02;

   $= \dfrac{\text{Total stress}}{39{,}468 \text{ psi}}$ for Node 005M

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated

4. Arbitrary intermediate breaks are not postulated.

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Main Feedwater System
(Inside Containment)

Problem No.: FW-03
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 2C)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2], [4]
N02	17,429	0.359	SG Nozzle	TE/C
005M	24,296	0.616	Elbow	(3)
110	16,374	0.505	Containment Penetration (M-8)	TE/C

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\,S_h + S_A)} = \frac{\text{Total stress}}{32{,}400 \text{ psi}}$ for Node 110;

   $= \frac{\text{Total stress}}{48{,}600 \text{ psi}}$ for N02;

   $= \frac{\text{Total stress}}{39{,}468 \text{ psi}}$ for 005M

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated

4. Arbitrary intermediate breaks are not postulated.

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Main Feedwater System
(Inside Containment)

Problem No.: FW-04
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 2H)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2)(4)
N02	24,763	0.627	SG Nozzle	TE/C
005M	24,785	0.628	Elbow	(3)
110	18,126	0.559	Containment Penetration (M-5)	TE/C

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\,S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$ for Node 110

$= \frac{\text{Total stress}}{39{,}468\ \text{psi}}$ for Nodes N02 & 005M

2. TE = Terminal End C = Circumferential

IM = Intermediate L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated

4. Arbitrary intermediate breaks are not postulated.

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Main Feedwater System
(Inside Containment)

Problem No.: FW-04
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 2D)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2), (4)
N02	21,543	0.546	SG Nozzle	TE/C
005M	23,609	0.598	Elbow	(3)
110	18,464	0.570	Containment Penetration (M-5)	TE/C

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$ for Node 110;

$$= \frac{\text{Total stress}}{39{,}468} \text{ for N02 \& 005M}$$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated

4. Arbitrary intermediate breaks are not postulated.

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Auxiliary Feedwater System
(Inside Containment)

Problem No.: AF-01
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 3E)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2)(4)
N04	17,895	0.473	SG Nozzle	TE/C
125	18,358	0.567	Containment Penetration (M-94)	TE/C
DD	29,415	0.908	8″ x 3″ Branch Connection	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\, S_h + S_A)} = \frac{\text{Total stress}}{32{,}400 \text{ psi}}$ for Nodes 125 & DD;

$= \frac{\text{Total stress}}{37{,}800 \text{ psi}}$ for Nodes N04

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated

4. Arbitrary intermediate breaks are not postulated.

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Auxiliary Feedwater System
(Inside Containment)

Problem No.: AF-01
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 3A)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2], [4]
N04	18,913	0.500	SG Nozzle	TE/C
125	19,226	0.593	Containment Penetration (M-94)	TE/C
DD	29,327	0.905	8″ x 3″ Branch Connection	(3)

1. Ratio = Total stress/stress limit $= \frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$ for Nodes 125 & DD;
   $= \frac{\text{Total stress}}{37{,}800\ \text{psi}}$ for N04
2. TE = Terminal End C = Circumferential
   IM = Intermediate L = Longitudinal
3. Highest relative stress point other than terminal ends: No break postulated
4. Arbitrary intermediate breaks are not postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Auxiliary Feedwater System (Inside Containment)

Problem No.: AF-02
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 3F)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2)(5)
N04	50,856	1.046	SG Nozzle	TE/C(3)
M95	19,657	0.607	Containment Penetration (M-95)	TE/C
57A	28,026	0.741	8"x 6" Reducer	(3), (4)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$ for Node M95;

   $= \frac{\text{Total stress}}{48{,}600\ \text{psi}}$ for Node N04;

   $= \frac{\text{Total stress}}{37{,}800\ \text{psi}}$ for Node 57A

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated
4. No break postulated
5. Arbitrary intermediate breaks are not postulated.

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Auxiliary Feedwater System
(Inside Containment)

Problem No.: AF-02
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 3B)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2), (4)
N04	49,454	1.018	SG Nozzle	TE/C(3)
M95	19,558	0.604	Containment Penetration (M-95)	TE/C
57A	27,669	0.732	8" x 6" Reducer	(3), (4)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8\,(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}} =$ for Node M95;

$= \frac{\text{Total stress}}{48{,}600\ \text{psi}}$ for Node N04;

$= \frac{\text{Total stress}}{37{,}800\ \text{psi}}$ for Node 57A

2. TE = Terminal End C = Circumferential

IM = Intermediate L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated
4. Arbitrary intermediate leaks are not postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Auxiliary Feedwater System (Inside Containment)

Problem No.: AF-03
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 3G)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2)(4)
N04	19,346	0.398	SG Nozzle	TE/C
M84	21,147	0.653	Containment Penetration (M-84)	TE/C
C5M	26,942	0.832	Elbow	(3)

1. Ratio = Total stress/stress limit = $\dfrac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \dfrac{\text{Total stress}}{32{,}400\ \text{psi}}$ for Nodes M84 & C5M;

   $= \dfrac{\text{Total stress}}{48{,}600\ \text{psi}}$ for Node N04;

2. TE = Terminal End   C = Circumferential

   IM = Intermediate   L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated

4. Arbitrary intermediate breaks are not postulated.

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Auxiliary Feedwater System (Inside Containment)

Problem No.: AF-03
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 3C)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2], [4]
N04	19,425	0.400	SG Nozzle	TE/C
M84	18,656	0.576	Containment Penetration (M-84)	TE/C
C35	26,572	0.820	8” x 3” Branch Connection Weld	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$ for Nodes M84 & C35

$$= \frac{\text{Total stress}}{48{,}600\ \text{psi}} \text{ for Node N04}$$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated

4. Arbitrary intermediate breaks are not postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Auxiliary Feedwater System (Inside Containment)

Problem No.: AF-04
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 3H)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2)(4)
N04	25,489	0.524	SG Nozzle	TE/C
150	14,715	0.454	Containment Penetration (M-83)	TE/C
178	24,369	0.752	8″ x 3″ Branch	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\,S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\text{ psi}}$ for Nodes 150 & 178;

   $= \frac{\text{Total stress}}{48{,}600\text{ psi}}$ for Node N04;

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated

4. Arbitrary intermediate breaks are not postulated.

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Auxiliary Feedwater System
(Inside Containment)

Problem No.: AF-04
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 3D)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2), (4)
N04	26,169	0.538	SG Nozzle	TE/C
150	16,573	0.512	Containment Penetration (M-83)	TE/C
178	23,847	0.736	8″ x 3″ Branch	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$ for Nodes 150 & 178;

$$= \frac{\text{Total stress}}{48{,}600\ \text{psi}} \text{ for Node N04}$$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point other than terminal ends: No break postulated

4. Arbitrary intermediate breaks are not postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Auxiliary Feedwater System (Outside Containment)

Problem No.: AF-11
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 4A)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
10	9,073	0.280	Containment Penetration (M-94)	TE/C
115	11,267	0.348	Normally Closed Valve	TE/C
315	12,946	0.400	Normally Closed Valve	TE/C
773	33,884	1.046	18″ FW Header Nozzle	TE/C
755	14,460	0.446	Normally Closed Valve	TE/C
42	29,125	0.899	8″ x 4″ Weldolet	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End   C = Circumferential

   IM = Intermediate   L = Longitudinal

3. Highest relative stress point other than break point 773

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Auxiliary Feedwater System
(Outside Containment)

Problem No.: AF-12
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 4B)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
10	7,423	0.229	Containment Penetration (M-95)	TE/C
75	17,531	0.541	Normally Closed valve	TE/C
315	16,245	0.501	Normally Closed valve	TE/C
822	27,082	0.836	18″ FW Header Nozzle	TE/C
795	12,589	0.389	Normally Closed valve	TE/C
298	28,480	0.879	Reducer	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End   C = Circumferential

   IM = Intermediate   L = Longitudinal

3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Auxiliary Feedwater System (Outside Containment)

Problem No.: AF-13
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 4C)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2)
10	7,914	0.244	Containment Penetration (M-84)	TE/C
75	23,693	0.731	Normally Closed valve	TE/C
425	27,955	0.863	Normally Closed valve	TE/C
376	32,407	1.000	18″ FW Header Nozzle	TE/C(3)
355	13,359	0.412	Normally Closed valve	TE/C

1. Ratio = Total stress/stress limit = $\dfrac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \dfrac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Auxiliary Feedwater System (Outside Containment)

Problem No.: AF-14
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 4D)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
10	7,336	0.226	Containment Penetration (M-83)	TE/C
79	34,527	1.066	Normally Closed valve	TE/C
315	19,646	0.606	Normally Closed valve	TE/C
795	19,517	0.602	Normally Closed valve	TE/C
826	5,074	0.157	18″ FW Header Nozzle	TE/C
70B	29,227	0.902	4″ Elbow	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point other than Node 79

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Steam Generator Blowdown System (Inside Containment)

Problem No.: SB-01
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 5A)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
1020	3,994	0.123	Containment Penetration (M-63)	TE/C
570	17,356	0.536	SG Nozzle	TE/C
2760	31,952	0.986	SG Nozzle	TE/C
1380	23,451	0.724	-	(3)

1. Ratio = Total stress/stress limit = $\dfrac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \dfrac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal Arbitrary Intermediate Breaks are not postulated.

3. Highest relative stress point other than the TE: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Steam Generator Blowdown System (Inside Containment)

Problem No.: SB-01
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 5A)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
4	5,008	0.155	Containment Penetration (M-63)	TE/C
31A	10,213	0.315	SG Nozzle	TE/C
83A	25,222	0.778	SG Nozzle	TE/C
69	27,331	0.844	-	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point of the piping system

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Steam Generator Blowdown System (Inside Containment)

Problem No.: SB-02
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 5B)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
1010	5,416	0.167	Containment Penetration (M-64)	TE/C
2230	30,736	0.949	SG Nozzle	TE/C
3200	17,331	0.535	SG Nozzle	TE/C
1410	28,325	0.874	-	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End   C = Circumferential

   IM = Intermediate   L = Longitudinal   Arbitrary Intermediate Breaks are not postulated.

3. Highest relative stress point other than TE: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Steam Generator Blowdown System (Inside Containment)

Problem No.: SB-02
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 5B)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2)
1	8,440	0.260	Containment Penetration (M-64)	TE/C
110	44,491	1.373	SG Nozzle	TE/C
142	15,629	0.482	SG Nozzle	TE/C
34	20,399	0.630	-	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End   C = Circumferential

   IM = Intermediate   L = Longitudinal

3. Highest relative stress point other than TE: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Steam Generator Blowdown System (Inside Containment)

Problem No.: SB-03
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 5C)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
1020	8,724	0.269	Containment Penetration (M-65)	TE/C
630	16,692	0.515	SG Nozzle	TE/C
3410	26,031	0.803	SG Nozzle	TE/C
1430	30,543	0.943	-	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal Arbitrary Intermediate Breaks are not postulated

3. Highest relative stress point other than TE: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Steam Generator Blowdown System
(Inside Containment)

Problem No.: SB-03
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 5C)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
3	10,003	0.309	Containment Penetration (M-65)	TE/C
99	19,540	0.603	SG Nozzle	TE/C
162	29,078	0.897	SG Nozzle	TE/C
101	24,853	0.767	-	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\,S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\text{ psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point other than TE: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Steam Generator Blowdown System
(Inside Containment)

Problem No.: SB-04
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 5D)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2)
1020	10,766	0.332	Containment Penetration (M-62)	TE/C
2030	23,854	0.736	SG Nozzle	TE/C
3370	29,062	0.897	SG Nozzle	TE/C
3020	20,772	0.641	Weld at Latrolet	IM/C(4)
2010	21,538	0.665	-	(3) (5)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\,S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\text{ psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal Arbitrary Intermediate Breaks are not postulated

3. Highest Relative Stress point other than TE: No break postulated.
4. Intermediate break previously postulated per RC-9827, Rev.4 (Previous Node 104)
5. Westinghouse Node number at the 2 ½” x 2” reducer in rerouted section of pipe.

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Steam Generator Blowdown System
(Inside Containment)

Problem No.: SB-04
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 5D)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
2	12,594	0.389	Containment Penetration (M-62)	TE/C
86	23,581	0.728	SG Nozzle	TE/C
150	31,997	0.988	SG Nozzle	TE/C
104	34,051	1.051	Weld at Latrolet	IM/C
149	25,897	0.799	-	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End   C = Circumferential

   IM = Intermediate   L = Longitudinal

3. Highest relative stress point other than Node 104: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Steam Generator Blowdown System (Inside Containment)

Problem No.: SB-01
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 5E)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
8030	10,441	0.322	Containment Penetration (M-63)	TE/C
1832	34,046	1.051	SG Nozzle	TE/C
562	22,960	0.709	SG Nozzle	TE/C
1450	31,981	0.987	-	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

   Arbitrary Intermediate Breaks are not Postulated.

3. Highest relative stress point of the piping system

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Steam Generator Blowdown System
(Inside Containment)

Problem No.: SB-02
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 5F)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
8015	8,857	0.273	Containment Penetration (M-64)	TE/C
1100	45,918	1.417	SG Nozzle	TE/C
2595	25,224	0.779	SG Nozzle	TE/C
1090	31,678	0.978	-	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

   Arbitrary Intermediate Breaks are not Postulated.

3. Highest relative stress point other than TE: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Steam Generator Blowdown System
(Inside Containment)

Problem No.: SB-03
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 5G)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2)
8020	11,263	0.348	Containment Penetration (M-65)	TE/C
625	24,316	0.750	SG Nozzle	TE/C
1620	31,355	0.968	SG Nozzle	TE/C
1010	31,127	0.961	-	(3)

1. Ratio = Total stress/stress limit = $\dfrac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \dfrac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

   Arbitrary Intermediate Breaks are not Postulated.

3. Highest relative stress point other than TE: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Steam Generator Blowdown System
(Inside Containment)

Problem No.: SB-04
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 5H)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
8010	13,328	0.411	Containment Penetration (M-62)	TE/C
3030	29,003	0.895	SG Nozzle	TE/C
1500	34,783	1.074	SG Nozzle	TE/C
1040	43,618	1.346	Weld at Latrolet	IM/C
3015	31,915	0.985	-	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

   Arbitrary Intermediate Breaks are not Postulated.

3. Highest relative stress point other than Node 1040 and TE: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Steam Generator Blowdown System
(Outside Containment)

Problem No.: SB-11
Revision: 2

UFSAR Figure: Figure 3.6.1-1 (Sheet 6A)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2)
E45	13,421	0.414	Containment Penetration (M-63)	TE/C
D84	9,549	0.295	Normally Closed Valve	TE/C
F31	12,405	0.383	Anchor	TE/C
E68	30,962	0.956	Valve	(3)
E73	32,982	1.18	Reducer at Valve FV-4157	IM/C

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400 \text{ psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Second highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Steam Generator Blowdown System
(Outside Containment)

Problem No.: SB-12
Revision: 2

UFSAR Figure: Figure 3.6.1-1 (Sheet 6B)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
E45	22,060	0.681	Containment Penetration (M-64)	TE/C
F31	15,849	0.489	Anchor	TE/C
D84	12,225	0.377	Normally Closed Valve	TE/C
P2	30,177	0.931	-	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End   C = Circumferential

   IM = Intermediate   L = Longitudinal

3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Steam Generator Blowdown System (Outside Containment)

Problem No.: SB-13
Revision: 2

UFSAR Figure: Figure 3.6.1-1 (Sheet 6C)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks(2)
M51	15,602	0.482	Containment Penetration (M-65)	TE/C
N58	11,461	0.354	Anchor	TE/C
L94	12,769	0.394	Normally Closed Valve	TE/C
M88	37,910	1.17	Reducer	IM/C,L
M94	40,546	1.25	Reducer	IM/C,L
M42	28,183	0.870	(3)	-

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End   C = Circumferential

   IM = Intermediate   L = Longitudinal

3. Highest relative stress point other than IM: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Steam Generator Blowdown System
(Outside Containment)

Problem No.: SB-14
Revision: 2

UFSAR Figure: Figure 3.6.1-1 (Sheet 6D)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks[2]
M51	13,163	0.406	Containment Penetration (M-62)	TE/C
N55	14,866	0.459	Anchor	TE/C
L95	8,106	0.250	Normally Closed Valve	TE/C
N34	23,694	0.731	-	(3)

1. Ratio = Total stress/stress limit = $\dfrac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \dfrac{\text{Total stress}}{32{,}400\ \text{psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Pressurizer Spray Line

Problem No.: RC-02
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 7B)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated[1]
5	17,896	-	-	40,080	0.0000	RCL Nozzle	TE/C
440	18,814	-	-	40,080	0.0000	RCL Nozzle	TE/C
605	73,613	50,810	25,972	40,680	0.5690	3″ x 2″ Reducer	IM/C
255	59,932	16,404	31,792	31,248	0.9532	Pressurizer Nozzle	TE/C
240E	16,750	-	-	39,456	0.6891	Elbow	IM/C,L
250B	16,234	-	-	39,456	0.7075	Elbow	IM/C,L
250E	17,311	-	-	39,456	0.7373	Elbow	IM/C,L

1. TE = Terminated C = Circumferential
   IM = Intermediate L = Longitudinal

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Reactor Coolant Loop 2 Drain to Reactor Coolant Drain Tank

Problem No.: RC-19
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 7C)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated[1]
20	46,698	1,811	28,969	40,080	0.0080	Crossover Leg Nozzle	TE/C
36	41,328	1,611	26,011	40,080	0.0091	Normally Closed Valve	TE/C

1. TE = Terminated; C = Circumferential; IM = Intermediate; L = Longitudinal

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Reactor Coolant Loop 2 Drain to Reactor Coolant Drain Tank

Problem No.: RC-19
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 7C)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated(1)
20	49,862	16,461	30,820	40,080	0.0236	Crossover Leg Nozzle	TE/C
36	47,380	18,630	26,550	40,080	0.0297	Normally Closed Valve	TE/C

1. TE = Terminated; C = Circumferential
   IM = Intermediate; L = Longitudinal

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Reactor Coolant Loop 2 Drain to Reactor Coolant Drain Tank

Problem No.: RC-20
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 7D)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated(1)
04	45,884	971	28,427	40,080	0.0070	Crossover Leg Nozzle	TE/C
14	39,196	-	-	40,080	0.0070	Normally Closed Valve	TE/C

1. TE = Terminated   C = Circumferential
   IM = Intermediate   L = Longitudinal

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Reactor Coolant Loop 2 Drain to Reactor Coolant Drain Tank

Problem No.: RC-20
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 7D)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated(1)
04	49,887	9,273	31,521	40,080	0.0012	Crossover Leg Nozzle	TE/C
14	41,519	2,625	24,362	40,080	0.0119	Normally Closed Valve	TE/C

1. TE = Terminated C = Circumferential
   IM = Intermediate L = Longitudinal

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Residual Heat Removal
(Pump A Suction: Inside
Containment)

Problem No.: RHR/SI-01
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 8A)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated[1]
135	28,018	-	-	39,456	0.0000	RC Nozzle	TE/C
86	65,954	1,202	52,986	43,200	0.7785	Normally Closed Valve	TE/C
130B	30,014	-	-	39,456	0.0014	Elbow	(2)
120E	30,006	-	-	39,456	0.0454	Elbow	(2)

1. TE = Terminated C = Circumferential

   IM = Intermediate L = Longitudinal

2. Highest relative stress and usage factor point other than TE: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Residual Heat Removal (Pump A Suction: Inside Containment)

Problem No.: RHR/SI-01
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 8A)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated[1]
135	27,983	-	-	39,456	0.0000	RC Nozzle	TE/C
86	65,860	2,227	52,891	43,200	0.811	Normally Closed Valve	TE/C
130E	28,316	-	-	39,456	0.0000	Elbow	(2)
120B	20,527	-	-	39,456	0.0318	Elbow	(2)

1. TE = Terminated C = Circumferential
   IM = Intermediate L = Longitudinal
2. Highest relative stress and usage factor point other than TE: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Residual Heat Removal (Pump B Suction: Inside Containment)

Problem No.: RHR/SI-09
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 8B)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated(1)
135	28,317	-	-	39,456	0.0000	RC Nozzle	TE/C
86	66,261	726	53,188	44,220	0.8163	Normally Closed Valve	TE/C
120E	32,389	-	-	39,456	0.0468	Elbow	(2)

1. TE = Terminated C = Circumferential
   IM = Intermediate L = Longitudinal
2. Highest relative stress and usage factor point other than TE: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Residual Heat Removal (Pump B Suction: Inside Containment)

Problem No.: RHR/SI-09
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 8B)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated[1]
135	30,345	-	-	39,456	0.0000	RC Nozzle	TE/C
86	66,745	726	53,672	44,220	0.8163	Normally Closed Valve	TE/C
120E	34,417	-	-	39,456	0.0468	Elbow	(2)

1. TE = Terminated C = Circumferential
   IM = Intermediate L = Longitudinal
2. Highest relative stress and usage factor point other than TE: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Residual Heat Removal (Pump C Suction: Inside Containment)

Problem No.: RHR/SI-16
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 8C)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated[(1)]
498	28,800	N/A	N/A	39,456	0.0000	RC Nozzle	TE/C
471C	67,361	2,980	53,044	43,200	0.8713	Normally Closed Valve	TE/C
495E	32,250	N/A	N/A	39,456	0.0474	Elbow	(2)

1. TE = Terminated C = Circumferential

   IM = Intermediate L = Longitudinal

2. Highest relative stress and usage factor point other than TE: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Safety Injection to Hot Leg
Loop 1: (Inside Containment)

Problem No.: RHR/SI-05
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 8D)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated[1]
440	24,303	-	-	38,592	0.0000	Hot Leg Nozzle	TE/C
430	39,122	2,918	33,276	38,592	0.0023	Valve End	TE/C

1. TE = Terminated C = Circumferential
   IM = Intermediate L = Longitudinal

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Safety Injection to Hot Leg
Loop 2: (Inside Containment)

Problem No.: RHR/SI-13
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 8E)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated[1]
453	26,182	-	-	38,592	0.0000	Hot Leg Nozzle	TE/C
448	40,063	4,175	33,675	38,592	0.0023	Valve End	TE/C

1. TE = Terminated   C = Circumferential
   IM = Intermediate   L = Longitudinal

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Safety Injection to Hot Leg
Loop 3: (Inside Containment)

Problem No.: RHR/SI-20
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 8F)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated[1]
960	19,943	-	-	39,509	0.0000	Hot Leg Nozzle	TE/C
956	36,340	-	-	39,264	0.0009	Valve End	TE/C

1. TE = Terminated   C = Circumferential
   IM = Intermediate   L = Longitudinal

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Accumulator Drain to the Reactor Coolant Drain Tank

Problem No.: RHR/SI-06, 14, 22
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 8J)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks (2)
140	12,784	0.318	Acc. Inj. A Line Nozzle	TE/C
139	10,993	0.274	Normally Closed Valve	TE/C
11	11,229	0.280	Acc. Inj. B Line Nozzle	TE/C
12	10,149	0.253	Normally Closed Valve	TE/C
89	15,532	0.387	Acc. Inj. C Line Nozzle	TE/C
70	10,112	0.252	Normally Closed Valve	TE/C

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{40{,}144\ \text{psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Chemical Volume and Control System Letdown from Cross Over Leg to Regenerative Heat Exchanger

Problem No.: CV-01
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 9A)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated[1]
142	21,864	N/A	N/A	33,542	0.0333	RCS Nozzle	TE/C
151	46,090	11,542	32,416	40,742	0.0965	Normally Closed Valve	TE/C
5A	(2)	-	-	N/A	N/A	Regenerative Heat Exchanger	TE/C

1. TE = Terminated　　C = Circumferential

   IM = Intermediate　　L = Longitudinal

2. Class 2 Piping: Total Stress = Equation 9 + Equation 10 Stress = 9,516 psi; $.8(1.2S_h + S_A)$ = 31,940 psi

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Chemical Volume and Control System Letdown from Cross Over Leg to Regenerative Heat Exchanger

Problem No.: CV-01
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 9A)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated[1]
142	33,340	N/A	N/A	39,322	0.0434	RCS Nozzle	TE/C
151	47,124	9,910	34,849	40,742	0.0946	Normally Closed Valve	TE/C
137	63,432	13,092	41,527	39,322	0.1503	4″ x 4″ x 2″ Tee	IM/C
5A	(2)	-	-	N/A	N/A	Regenerative Heat Exchanger	TE/C

1. TE = Terminated C = Circumferential

   IM = Intermediate L = Longitudinal

2. Class 2 Piping: Total Stress = Equation 9 + Equation 10 Stress = 9,516 psi; $.8(1.2S_h + S_A) = 31,940$ psi

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 9B)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
815	7,806	0.241	Containment Penetration (M-46)	TE/C
500	8,997	0.277	Letdown Heat Exchanger Nozzle	TE/C
604	10,375	0.320	Normally Closed Valve	TE/C
581	16,793	0.518	Normally Closed Valve	TE/C
50	7,795	0.240	Anchor HL5005	TE/C
315	9,902	0.305	Anchor HL5038	TE/C
575	18,989	0.585	-	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{32{,}439\ \text{psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control Normal Charging Inside Containment

Problem No.: CV-05
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 9C)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated[1]
1	20,620	-	-	38,880	0.0264	Cold Leg Nozzle	TE/C
123	(2)	-	-	-	-	Regenerative Heat Exchanger Nozzle	TE/C

1. TE = Terminated C = Circumferential

   IM = Intermediate L = Longitudinal

2. Class 2 Piping: Total Stress = Equation 9 + Equation 10 Stress = 8,366 psi; $.8(1.2S_h + S_A) = 32,091$ psi

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control
Auxiliary Pressurizer Spray

Problem No.: CV-07
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 9D)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated(2)
800	(1)	-	-	-	-	Class 2 Anchor	TE/C
607	67,603	42,807	28,059	39,456	0.308	Check Valve	IM/C

2. Class 2 Piping: Total Stress = Equation 9 + Equation 10 Stress = 6,228 psi; $.8(1.2S_h + S_A)$ = 30,803 psi

1. TE = Terminated C = Circumferential

IM = Intermediate L = Longitudinal

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Chemical Volume and Control Alternate Charging Line to Loop 3

Problem No.: CV-06
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 9E)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	2.4S_m (psi)	Usage Factor	Location	Type of Break Postulated(1)
261	23,969	-	-	38,880	0.0312	Cold Leg Nozzle	TE/C
359	(2)	-	-	-	-	Anchor	TE/C

1. TE = Terminated C = Circumferential

   IM = Intermediate L = Longitudinal

2. Class 2 Piping: Total Stress = Equation 9 + Equation 10 Stress = 19,308 psi; .8(1.2S_h + S_A) = 32,091 psi

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Chemical Volume and Control Alternate Charging Line to Loop 3

Problem No.: CV-06
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 9E)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated(1)
261	23,969	-	-	38,880	0.0312	Cold Leg Nozzle	TE/C
359	(2)			-	-	Anchor	TE/C

1. TE = Terminated; C = Circumferential; IM = Intermediate; L = Longitudinal

2. Class 2 Piping: Total Stress = Equation 9 + Equation 10 Stress = 36,683 psi; $.8(1.2S_h + S_A)$ = 32,091 psi

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control
Excess Letdown from Loop 4

Problem No.: CV-17
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 9F)

Node	Eq. 10 Stress (psi)	Eq. 12 Stress (psi)	Eq. 13 Stress (psi)	$2.4S_m$ (psi)	Usage Factor	Location	Type of Break Postulated(1)
240	55,180	18,570	36,179	40,742	0.0281	Cold Leg Nozzle	TE/C
252	47,242	7,141	28,833	40,742	0.0192	Normally Closed Valve	TE/C
1	-	-	-	-	-	Heat Exchanger Nozzle	TE/C(1)
226	53,542	8,177	35,159	40,742	0.0056	Tee	(3)

1. Class 2 Piping: Total Stress = Equation 9 + Equation 10 Stress = 4,971 psi; $.8(1.2S_h + S_A)$ = 31,673 psi
2. TE = Terminated C = Circumferential

   IM = Intermediate L = Longitudinal
3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control Excess Letdown to Heat Exchanger

Problem No.: CV-19
Revision: 5

UFSAR Figure: Figure 3.6.1-1 (Sheet 9G)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
5	8,835	0.261	Heat Exchanger Nozzle	TE/C
45	7,777	0.229	2″ x 1″ Reducer	TE/C

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{33{,}912\ \text{psi}}$

2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 1 ONLY)

System: Chemical Volume and Control
Excess Letdown to
Containment Penetration

Problem No.: CV-02/03
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 9H)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks (2)
8	14,441	0.436	Containment Penetration (M-46)	TE/C
248	9,737	0.294	Regenerative Heat Exchanger Outlet	TE/C
94	25,050	0.756	Normally Closed Valve	TE/C
260	12,495	0.377	Normally Closed Valve	TE/C
172	18,329	0.553	Anchor HL5038	TE/C
284	10,402	0.632(4)	Valve End	TE/C
64	29,250	0.883	-	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{33{,}135\ \text{psi}}$
2. TE = Terminal End   C = Circumferential

   IM = Intermediate   L = Longitudinal
3. Highest relative stress point: No break postulated
4. $.8\ (1.2S_h+S_A) = 16{,}446$ psi

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

(FOR UNIT 2 ONLY)

System: Chemical Volume and Control Excess Letdown to Containment Penetration

Problem No.: CV-02/03
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 9H)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks (2)
8	13,488	0.407	Containment Penetration (M-46)	TE/C
248	6,749	0.204	Regenerative Heat Exchanger Outlet	TE/C
94	25,043	0.756	Normally Closed Valve	TE/C
260	11,297	0.341	Normally Closed Valve	TE/C
172	16,661	0.503	Anchor HL5038	TE/C
284	10,402	0.632(4)	Valve End	TE/C
64	29,706	0.896	-	(3)

1. Ratio = Total stress/stress limit = $\frac{\text{Total stress}}{.8(1.2\ S_h + S_A)} = \frac{\text{Total stress}}{33{,}135\ \text{psi}}$
2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal
3. Highest relative stress point: No break postulated
4. $.8\ (1.2S_h+S_A) = 16{,}446$ psi

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10A)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
315	9,576	0.295	Anchor	TE/C
445	9,313	0.287	Letdown Reheat Heat Exchanger 1A	TE/C
375	22,451	0.692	-	(3)

1. Ratio = Total stress/stress limit = .8 (1.2 $S_h + S_A$) = 32,439 psi
2. TE = Terminal End C = Circumferential
   IM = Intermediate L = Longitudinal
3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10B)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
50	11,637	0.354	Anchor	TE/C
200	5,947	0.181	Letdown Reheat Heat Exchanger 1A	TE/C
100B	17,368	0.528	Elbow	(3)

1. Ratio = Total stress/stress limit = $.8(1.2\ S_h + S_A)$ = 32,891 psi
2. TE = Terminal End   C = Circumferential
   IM = Intermediate   L = Longitudinal
3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10C)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
25	6,193	0.366	Normally Closed Valve	TE/C
385	4,444	0.262	R.C. Filter 1A	TE/C
720	5,633	0.322	Anchor	TE/C
775	3,060	0.180	Normally Closed Valve	TE/C
905	2,726	0.161	Normally Closed Valve	TE/C
999	2,961	0.175	Anchor	TE/C
525	8,467	0.500	-	(3)

1. Ratio = Total stress/stress limit = .8(1.2 S_h + S_A) = 16,956 psi
2. TE = Terminal End   C = Circumferential

   IM = Intermediate   L = Longitudinal
3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10D)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
1B	6,917	0.204	Letdown Heat Exchanger 1A	TE/C
200	3,878	0.114	Anchor	TE/C
250	14,676	0.433	Normally Closed Valve	TE/C
115	14,632	0.432	-	(3)

1. Ratio = Total stress/stress limit = .8(1.2 S_h + S_A) = 33,912 psi
2. TE = Terminal End C = Circumferential
   IM = Intermediate L = Longitudinal
3. Highest relative stress point in high-energy region: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 3

UFSAR Figure: Figure 3.6.1-1 (Sheet 10E)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
5	6,271	0.185	Cation Bed Demineralizer 1B & 1A	TE/C
35	893	0.012	Normally Closed Valves (CV-119A & B, CV-128A & B)	TE/C
5	5,745	0.169	Mixed Bed Demineralizer 1A & 1B	TE/C
15E	9,872	0.291	Elbow	(3)

1. Ratio = Total stress/stress limit = .8(1.2 S_h + S_A) = 33,912 psi
2. TE = Terminal End   C = Circumferential
   IM = Intermediate   L = Longitudinal
3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10F)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks (2)
100	2,943	0.073	Anchor HL5003	TE/C
250	2,743	0.068	Mixed Bed Demineralizer 1A	TE/C
390	3,798	0.094	Mixed Bed Demineralizer 1B	TE/C
505	4,924	0.122	Letdown Filter 1B	TE/C
605	1,458	0.036	Letdown Filter 1B	TE/C
640	2,600	0.064	Anchor HL5006	TE/C
260	6,448	0.159	-	(3)

1. Ratio = Total stress/stress limit = .8(1.2 S_h + S_A) = 40,434 psi
2. TE = Terminal End C = Circumferential
   IM = Intermediate L = Longitudinal
3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10G)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
5	3,820	0.113	Letdown Filter 1A	TE/C
100	2,800	0.083	Anchor HL5003	TE/C
314	3,491	0.103	Cation Bed Demineralizer 1B	TE/C
242	3,616	0.107	Cation Bed Demineralizer 1A	TE/C
460	2,178	0.064	Cation Bed Demineralizer 1A	TE/C
426	3,175	0.094	Anchor HL5002	TE/C
494	2,130	0.063	Cation Bed Demineralizer 1B	TE/C

1. Ratio = Total stress/stress limit = $.8(1.2 S_h + S_A)$ = 33,912 psi
2. TE = Terminal End C = Circumferential
   IM = Intermediate L = Longitudinal
3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10G)

Node	Total Stress (psi)	Ratio[(1)]	Location	Type of Postulated Breaks [(2)]
526	2,145	0.063	Mixed Bed Demineralizer 1A	TE/C
564	2,106	0.062	Mixed Bed Demineralizer 1B	TE/C
690	2,641	0.093[(4)]	Moderate Heat Exchanger 1A	TE/C
679	3,044	0.107[(4)]	Anchor HL5007	TE/C
574	7,674	0.226	-	(3)

1. Ratio = Total stress/stress limit = $.8(1.2\ S_h + S_A) = 33{,}912$ psi
2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal
3. Highest relative stress point: No break postulated
4. $.8(1.2S_h + S_A) = 28{,}363$,psi

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10H)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
175	1,875	0.055	Letdown Filter 1A	TE/C
195	3,480	0.103	Anchor AN1014	TE/C
300	2,656	0.078	Anchor HL5005	TE/C
640	3,218	0.095	Anchor AN1001	TE/C
74	5,992	0.177	-	(3)

1. Ratio = Total stress/stress limit = .8(1.2 S_h + S_A) = 33,912 psi
2. TE = Terminal End C = Circumferential
   IM = Intermediate L = Longitudinal
3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10I)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
568	1,820	0.054	Normally Closed Valve	TE/C
572	4,066	0.120	Large Pipe Connection	TE/C
584	2,458	0.072	Normally Closed Valve	TE/C
588	3,555	0.105	Letdown Pipe Connection	TE/C
540	10,840	0.320	-	(3)

1. Ratio = Total stress/stress limit = $.8(1.2\ S_h + S_A) = 33{,}912$ psi
2. TE = Terminal End C = Circumferential
   IM = Intermediate L = Longitudinal
3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10J)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
112	4,095	0.242	Volume Control Tank 1A	TE/C
114	5,320	0.314	Normally Closed Valve	TE/C
590	5,128	0.302	Anchor HL5001	TE/C
88	6,557	0.387	Normally Closed Valve	TE/C
40	12,918	0.761	-	(3)

1. Ratio = Total stress/stress limit; stress limit = $.8(1.2\ S_h + S_A) = 16{,}956$ psi
2. TE = Terminal End C = Circumferential
   IM = Intermediate L = Longitudinal
3. Highest relative stress point: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10K)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
5	4,653	0.328[4]	Anchor HL5003	TE/C
15	5,135	0.362[4]	Normally Closed Valve	TE/C
500	7,341	0.433	Anchor HL5012	TE/C
405	2,294	0.135	R.C. Filter 1A	TE/C
410	5,070	0.300	Anchor HL5005	TE/C
565	5,096	0.301	Anchor HL5002	TE/C
275	1,156	0.068	-	(3)

1. Ratio = Total stress/stress limit; stress limit = $.8(1.2\ S_h + S_A) = 16{,}956$ psi
2. TE = Terminal End   C = Circumferential

   IM = Intermediate   L = Longitudinal
3. Highest relative stress point other than TEs: No break postulated
4. $.8(1.2S_h + S_A) = 14{,}190$ psi

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10L)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks (2)
279	1,338	0.079	Mixed Bed Demineralizer 1A	TE/C
272	2,047	0.121	Normally Closed Valve	TE/C
173	879	0.052	Mixed Bed Demineralizer 1B	TE/C
166	149	0.012	Normally Closed Valve	TE/C
387	1,183	0.070	Cation Bed Demineralizer 1B	TE/C
379	1,463	0.086	Normally Closed Valve	TE/C
491	1,134	0.067	Cation Bed Demineralizer 1A	TE/C
484	1,369	0.081	Normally Closed Valve	TE/C

1. Ratio = Total stress/stress limit = $.8(1.2\ S_h + S_A)$ = 16,956 psi
2. TE = Terminal End   C = Circumferential
   IM = Intermediate   L = Longitudinal

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10L)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
304	4,704	0.277	Anchor HS5001	TE/C
305	2,832	0.167	Normally Closed Valve	TE/C
409	3,808	0.225	Anchor HS5001	TE/C
405	2,458	0.145	Normally Closed Valve	TE/C
207	5,751	0.339	Anchor HS5001	TE/C
205	3,206	0.189	Normally Closed Valve	TE/C
190	10,406	0.614	Anchor HS5001	TE/C[3]
184	6,359	0.375	Normally Closed Valve	TE/C

1. Ratio = Total stress/stress limit; stress limit = $.8(1.2\ S_h + S_A) = 16{,}956$ psi
2. TE = Terminal End C = Circumferential
   IM = Intermediate L = Longitudinal
3. Highest relative stress point

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Letdown Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10M)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks (2)
5	13,743	0.405	Centrifugal Charging Pump 1A	TE/C
135	11,128	0.328	Anchor	TE/C
166	26,134	0.771	Anchor	TE/C
190	28,130	0.830	Anchor	TE/C
225	18,713	0.552	Anchor	TE/C
370	5,538	0.163	Centrifugal Charging Pump 1B	TE/C
410	7,681	0.226	Anchor	TE/C
430	6,202	0.183	Anchor	TE/C
165	39,694	1.171	Tee	IM/C(4)
175	30,658	0.910	Valve End	(3)

1. Ratio = Total stress/stress limit; stress limit = .8(1.2 S_h + S_A) = 33,912 psi
2. TE = Terminal End C = Circumferential

   IM = Intermediate L = Longitudinal
3. Highest relative stress point
4. Effects from this break are envelope by break at Node 166

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control (Charging Outside Containment)

Problem No.: CV-11L
Revision: 4

UFSAR Figure: Figure 3.6.1-1 (Sheet 10N)

Node	Total Stress (psi)	Ratio(1)	Location	Type of Postulated Breaks (2)
130	11,308	0.333	Anchor	TE/C
175	6,588	0.194	Containment Penetration (M-48)	TE/C
174	8,717	0.257	-	(3)

1. Ratio = Total stress/stress limit = .8(1.2 S_h + S_A) = 33,912 psi
2. TE = Terminal End C = Circumferential
   IM = Intermediate L = Longitudinal
3. Highest relative stress point other than TE in high energy region: No break postulated

TABLE 3.6.2-1 (Continued)

HIGH-ENERGY PIPE BREAK STRESS SUMMARY RESULTS

System: Chemical Volume and Control
Water Seal Injection Line from RCP A, B, C, and D to Containment Penetration M51 and M52
(Inside Containment)

Problem No.: CV03
Revision: 0

UFSAR Figure: Figure 3.6.1-1 (Sheet 10P)

Node	Total Stress (psi)	Ratio[1]	Location	Type of Postulated Breaks [2]
390	22,084	0.651	Line A to Containment Penetration (M-51)	TE/C
5	10,017	0.295	RCP 1-A Seal Water Injection Nozzle	TE/C
29E	24,175	0.713	Elbow	(3)
777	12,090	0.357	RCP 1-B Seal Water Injection Nozzle	TE/C
95	13,243	0.391	Line B to Containment Penetration (M-51)	TE/C
60M	19,722	0.582	Bend	(3)
2	10,325	0.305	RCP 1-C Seal Water Injection Nozzle	TE/C
95	9,608	0.283	Line C to Containment Penetration (M-52)	TE/C
1A	11,572	0.341	RCP 1-D Seal Water Injection Nozzle	TE/C
95	14,376	0.424	Line D to Containment Penetration (M-52)	TE/C

1. Ratio = Total stress/33912 psi.
2. TE = Terminal End C = Circumferential
3. Highest relative stress point (class 2 piping): No break postulated
   Total stress = Equation 9 + Equation 10 = 0.8 (1.2 S_h +S_a) = 33,912

APPENDIX 3.6.A

ISOLATION VALVE CUBICLE SUBCOMPARTMENT ANALYSIS

3.6.A.1 Design Features.

The Isolation Valve Cubicle (IVC) is located between the Containment and Turbine Generator Building (TGB) on the north side of the Containment. The general arrangement drawings listed as Figures 1.2-21 through 1.2-25 in Table 1.2-1 provide the plan and elevation views of this area. The IVC consists of four cubicles with each cubicle designed to accommodate equipment and piping pertaining to each of the four trains of the steam and feedwater systems, thus meeting the train separation criteria.

At lower levels (between El. 10 ft-0 in. and 34 ft-0 in.) each train has an auxiliary feedwater (AFW) pump. Three of the pumps are motor-driven while the fourth is turbine-driven. Watertight doors assure the separability of the AFW pump cubicles from one another in the event of flooding of any one of the cubicles due to a pipe break. Main steam (MS) and main feedwater (FW) pipes run through the IVC above El. 34 ft-0 in. extending from the Containment penetrations to the five-way bending-torsional restraints mounted between two walls on the north end of the IVC. The main steam isolation valve (MSIV), MS safety valves, and main feedwater isolation valve (MFIV) are located in this compartment. A sloped metal roof covers the top of the IVC. The roof will lift off the affected cubicle in the event of a pressure build-up due to a pipe break in one of the cubicles. The AFW pump cubicles relieve their pressure build-up in the event of a AFW pipe break through the opening at El. 34 ft-0 in. from whence it is eventually vented to the atmosphere via the roof in the IVC.

3.6.A.2 Design Evaluation.

The MS and FW piping in this compartment is designed to the break exclusion criteria, stated in Section 3.6.2.1, for those portions of the piping passing through the primary containment and extending to the first pipe whip restraint past the first outside isolation valve. Accordingly, mechanistic pipe breaks are not postulated in the MSIV/MFIV piping. However, to provide an additional level of assurance of operability of safety-related equipment in this compartment, the building structures and safety-related equipment are designed to environmental conditions (pressure temperature and flooding) that would result from a break equal to one cross-sectional area of the MS and main FW piping. Adequate venting is provided to limit the pressurization of the cubicles to below the design pressure of the wall.

The following cases were analyzed to determine the worst environmental conditions for the IVC.

1. Main steam line break (MSLB) equivalent to the area of a single area rupture

2. Main FW line break due to a single area break

3. AFW line double-ended break in the AFW cubicle

4. Double-ended steam generator (SG) blowdown line break in common corridor area

In general, the calculated maximum pressures resulting from an MSLB are greater than those calculated for the other postulated break types. Therefore, only the MSLB results are presented. There is one exception. A break in the SG blowdown line results in the highest pressure calculated for the common corridor area north of the AFW pump cubicles at El. 10 ft. CN-2822

The MSLB subcompartment pressure analysis was performed using the GOTHIC 4.0 HLP-001 (Reference 3.6.A-8) computer code. Details of the code are given in Section 3.6.A.6. The short-term mass and energy releases for the IVC subcompartment pressure analysis are listed in Table 3.6.A-1. The mass and energy releases were determined using the RETRAN-03 computer code, which is discussed in Section 6.2.1.4.7.

For the MSLB analysis, the nodalization scheme is presented in Figure 3.6.A-1. The common corridor area is not part of this model. The nodal boundaries have been selected wherever there are flow restrictions (such as grating platforms). The roof of the IVC is covered by built-up metal panels. The differential pressures at which these panels lift is 0.8 psig. The weight of these panels is 3 lb/ft^2. The panel is assumed to move parallel to its original position (note the panel has a small slope away from the Containment Building) until it clears the sidewalls of the IVC. Once the panel clears the walls, it is assumed to lift away from the path of the flow of the steam-air mixture to the atmosphere. Thus, this movement of the panel above its nominal position creates movable nodes 8 and 9 shown in Figure 3.6.A-1. The position of the movable panels is modeled as a function of IVC pressure using the STEM_TRAVEL code (Reference 3.6.A-9). The node and junction parameters of the IVC are given on Table 3.6.A-2. The vent area and the volume of these nodes are given in Tables 3.6.A-3 and 3.6.A-4. CN-2822

Results of the cases which yield maximum pressures in the various nodes of an IVC cubicle including the associated AFW pump room are presented in Figure 3.6.A-3. In MSLB case 1, the mass and energy release is assigned to node 6, while in MSLB case 2, the mass and energy release goes to node 7. The peak pressures for the limiting case in each node are indicated in Table 3.6.A-2.

The pressure analyses for the steam generator blowdown line break in the common corridor area were modeled using the COPDA computer code. Details of the code are given in Section 6.2.1.2.3. Short-term mass and energy releases were calculated using the methodology of References 3.6-9, 3.6.A-6, and 3.6.A-7.

The nodalization model selected for the common corridor area is shown in Figure 3.6.A-5. The node and junction parameters of the common corridor area of the IVC are given in Table 3.6.A-6.

Peak pressures for the SG blowdown line break in the common corridor area of the IVC structure are presented in Table 3.6.A-6.

For generating the equipment qualification temperatures and pressures of the IVC, a simpler three-node model of the IVC has been used and the volume and junction properties were input into a modified COPDA code named FLUD (see Section 3.6.A.3 for discussion of FLUD). The simplified model consists of three nodes with node 1 being the AFW pump room between El. 10 ft and 32 ft, node 2 is between El. 34 ft and 55 ft-6 in., and node 3 occupying space above 55.5 ft. Out of the various cases considered, MSLB produced the limiting temperatures and pressures in the IVC. The long-term mass and energy release used in the analysis is presented in Table 3.6.A-5 and the temperature profiles are given in Figure 3.6.A-4. The mass and energy release has been obtained using Westinghouse LOFTRAN code (Ref. 3.6.A-5).

3.6.A.3 FLUD, a Compartment Differential Pressure Analysis Code.

This section describes the computational procedure and the analytical techniques used in FLUD. The analytical basis for COPDA is described in Reference 3.6.A-4. The set-up of initial conditions, the determination of the thermodynamic state point at subsequent time increments, and computation of energy and mass transport between one time step is discussed in Sections 3.6.A.3.1, 3.6.A.3.2, and 3.6.A.3.3 for FLUD. Selection was made of the control volume and flow path configuration that resulted in the best representation of the pressure transients in the compartments along the flow paths from the break. The major differences between FLUD and COPDA are the use of steam table curve fits (Section 3.6.A.3) instead of table look-ups, the equation of state, which is a first-order virial expansion (discussed in Section 3.6.A.3.1), and the capability of wall heat transfer calculation. The fluid flow equations (compressible equations, HEM model, and integrated momentum equation) used in COPDA have been reproduced in the FLUD code. It may be observed from the FLUD flowchart in Figure 3.6.A-2 that the calculational procedures for FLUD and COPDA are very similar.

3.6.A.3.1 Equation of State – This section describes how FLUD determines the thermodynamic state for each compartment in a system of interconnected compartments.

The thermodynamic system (compartment) is assumed to be in equilibrium. The states assumed by the air-steam-water mixture can be described in terms of thermodynamic coordinates, P, V, and T referring to the mixture as a whole. The equation of state is derived from a first order virial expansion as presented in Reference 3.6.A-1. Using the molecular theory of gases, the following equation of state for an air-steam mixture is obtained assuming negligible air-steam molecular interaction:

$$P = (M_a R_a + M_s R_s)\ \frac{T}{V} + \frac{(M_s)^2}{V} R_S T B_s(T),\ \ (\mathrm{lbf/ft^2})$$

(Eq. 3.6.A-1)

where the temperature dependence of the second virial coefficient for steam R_s (T) is given by (Ref. 3.6.A-2).

$$B_S(T) = 0.0330 - \frac{75.3137}{T} 10^{3.2659/(T^2\ \mathrm{x}\ 10^{-5} + 1.1308)}$$

(Eq. 3.6.A-2)

Equation 3.6.A-1 can be rewritten as the sum of the partial pressure of air P_a and the partial pressure of steam P_s where

$$P_a = \frac{M_a}{v}\ R_a T,\ (\mathrm{lbf/ft^2}) = 0.37043\ \frac{T,}{v_a}\ (\mathrm{psia})$$

(Eq. 3.6.A-3)

and

$$P_s = \frac{M_s}{V} R_s T[1 + \frac{M_s B_s}{V}(T)], \quad (lbf/ft^2) \qquad \text{(Eq. 3.6.A-4)}$$

Equation 3.6.A-4 compares well with the steam tables (Ref. 3.6.A-2). For example, the relative error in Eq. 3.6.A-4 is less than one percent for saturated steam at temperature 570°F.

3.6.A.3.2 Compartment Thermodynamic State – At any time, the total internal energy E, the air mass M_a, and the vapor mass M_v have known values for each compartment. Vapor is defined as a homogeneous mixture of steam and water in unknown proportions.

The internal energy is a function of as many thermodynamic coordinates as are necessary to specify the state of the system. Therefore, for known air and vapor masses, and because the compartment volume is originally specified, the compartment internal energy can be expressed as a function of temperature only:

$$E = E(T) \qquad \text{(Eq. 3.6.A-5)}$$

At the saturation temperature T_o, there is a discontinuous change in the slope of E (T) due to phase change in the compartment atmosphere. Associated with T_o is the compartment saturation energy $E_o = E(T_o)$. Equation 3.6.A-5 has two branches: (1) a two-phase branch where $E<E_o$ and $T<T_o$ and (2) a superheat branch where $E>E_o$ and $T>T_o$. Along the two-phase branch, the vapor portion of the atmosphere has a non-zero water mass component, while along the superheat branch the vapor contains no water.

Having examined the behavior of E(T), Equation 3.6.A-5 is solved for the compartment temperature, E being known. Parameters v_{sat}, e_{sat} and v_w, e_w represent the specific volumes and specific internal energies of saturated steam and water, respectively. The dependence of these quantities on temperature is determined empirically from steam table curve fits described in Section 3.6.A.5. E_o is calculated to determine on which branch of E(T) the compartment temperature lies. At compartment saturation, the steam mass M_s is identical to M_v and the specific volume of the steam is just $v_{sat}(T_o)$. Thus,

$$V = M_v v_{sat}(T_o) \qquad \text{(Eq. 3.6.A-6)}$$

The above equation is easily solved for T_o by utilizing the inverse of the function $v_{sat}(T_o)$, which is also a steam table curve fit where $T_o = T_{sat}(V/M_v)$. The saturation internal energy for the compartment is then given by

$$E_o = M_a c_{va} T_o + M_v e_{sat}(T_o) \qquad \text{(Eq. 3.6.A-7)}$$

where c_{va} = 0.1725 Btu/lbm-°R is the specific heat at constant volume for air averaged over the temperature range from – 109.7 to 440.3°F. For the case $E<E_o$ (the two-phase branch), the explicit dependence of E on M_a, M_s, M_w and T is

$$E = M_a c_{va} T + M_s(T) e_{sat}(T) + M_w(T) e_w(T) \qquad \text{(Eq. 3.6.A-8)}$$

The functions e_s (P_s, T) and e_w (T) are the specific internal energies of steam and water, respectively, and are also discussed in Section 3.6.A-5. The steam and water masses are functions of temperature only and are given by

$$M_s(T) = x(T)M_v = \frac{V - M_v v_w(T)}{v_{sat}(T) - v_w(T)} \qquad \text{(Eq. 3.6.A-9A)}$$

and

$$M_w(T) = M_v - M_s(T) \qquad \text{(Eq. 3.6.A-9B)}$$

where the steam quality x (T) is defined by the following:

$$x\,(T) = \frac{M_s(T)}{M_V} = \frac{V/M_v - v_w(T)}{v_{sat}(T) - v_w(T)} \qquad \text{(Eq. 3.6.A-10)}$$

For the case $E>E_o$ (the superheat branch), the explicit dependence of E is given by

$$E = M_a c_{va} T + M_s e_s(P_s, T) \qquad \text{(Eq. 3.6.A-11)}$$

The steam mass M_s is not a function of temperature since it is equal to the vapor mass M_v, and of course the water mass is zero.

Because E is a complex function of T as seen by the above, Equation 3.6.A-5 does not readily lend itself to a strictly analytical solution. Instead, FLUD employs a one-pass iterative technique to solve for the temperature.

3.6.A.3.3 Compartment Initial Conditions – The initial thermodynamic state is specified for each compartment by the total compartment pressure P, the compartment volume V, temperature T, relative humidity ϕ, and vapor quality x.

If $\phi < 1.0$, the compartment is superheated, the vapor consists entirely of steam, and the steam mass is given by definition as

$$M_s = \phi \frac{V}{v_{sat}(T)} \qquad \text{(Eq. 3.6.A-12)}$$

The steam partial pressure is obtained from Equation 3.6.A-4, and thus the air mass is given by Equation 3.6.A-3. The internal energy is calculated using Equation 3.6.A-11. If $\phi = 1.0$ and $x = 1.0$, the compartment is saturated. The steam partial pressure is given by the saturation pressure $P_s = P_{sat}(T)$. The saturation pressure of steam P_{sat} is obtained empirically from a curve fit to the steam tables. The steam mass is given by Equation 3.6.A-12 with $\phi = 1.0$. The vapor mass is identically equal to the steam mass, and the internal energy is computed for Equation 3.6.A-7. For ϕ =1.0 and $x < 1.0$, the compartment is two-phase. The vapor and steam masses are given by Equation 3.6.A-9A and the water mass by Equation 3.6.A-9B. The steam partial pressure is equal to the saturation pressure $P = P_{sat}$ (T). Therefore, the air mass can be calculated from Equation 3.6.A-3. However, because the compartment now contains water, the volume accessible to the air and steam V_g is just

$$V_g = V - M_w v_{sat}(T) \qquad \text{(Eq. 3.6.A-13)}$$

This gas volume V_g must be used in place of V in Equation 3.6.A-3 determining the air mass. The internal energy is obtained from Equation 3.6.A-8.

3.6.A.3.4 Air and Vapor Component Flow Rates – The time-dependent partial pressure of steam is given by Equation 3.6.A-4 where v_s replace V/M_s. The time-dependent air specific volume v_a is then obtained from Equation 3.6.A-3. Time-dependent air and steam mass fractions are then calculated as follows:

$$f_a = v_s / (v_s + v_a) \qquad \text{(Eq. 3.6.A-14A)}$$

$$f_v = v_a / (v_s + v_a) \qquad \text{(Eq. 3.6.A-14B)}$$

The flow rates of the air and vapor components that comprise the gas are calculated from the total flow rate M by using the mass fractions of air and vapor in the upstream compartment:

$$\dot{M}_{aij} = f_a M_{ij} \qquad \text{(Eq. 3.6.A-15)}$$

$$\dot{M}_{vij} = f_v M_{ij} \qquad \text{(Eq. 3.6.A-16)}$$

3.6.A.4 Energy Transfer Mechanisms.

There are several mechanisms by which FLUD transfers energy to and from the various compartments and the atmosphere. These mechanisms are:

1. Blowdown energy

2. Flow of energy between compartments

3. Compartment heat loads

4. Compartment unit coolers

All of these mechanisms add or subtract energy from the system. A continuous accounting of all energy contributors is kept by FLUD in the form of an overall energy balance to ensure energy conservation. The various energy transfer mechanisms and the energy balance are discussed below.

3.6.A.4.1 Blowdown Energy – Blowdown energy is added to the system of compartments when FLUD is used to analyze a high-energy pipe break problem. The blowdown flow rate M_B, specific enthalpy h_B, and the split among compartments are assumed to be given at input data. The rate of energy addition to the system by blowdown H_B is usually a time-varying quantity given by

$$\dot{H}_B = \dot{M}_B \ h_B \qquad \text{(Eq. 3.6.A-17)}$$

This variable energy rate is used to calculate the amount of energy that is placed in one or in the various break compartments during each time step. The total amount of blowdown energy added to the system is the integral of H_B

$$H_B \ (t) = \int_o^t \dot{H}_B \ dt \qquad \text{(Eq. 3.6.A-18)}$$

The blowdown energy rate added to the i^{th} compartment is calculated is multiplying the user-supplied split fraction for the i^{th} compartment times the total blowdown energy rate in Equation 3.6.A-17.

3.6.A.4.2 Enthalpy Flow – Whenever mass is transferred between compartments or between a compartment and the atmosphere, there is and associated transfer of energy based upon the enthalpy of the upstream compartment. The general relation used to calculate enthalpy flow between compartments is

$$\dot{H}_i = \sum_j \dot{M}_{ij} h^*_{ij} \qquad \text{(Eq. 3.6.A-19)}$$

where h^*_{ij} represents the total specific enthalpy of the gas in the upstream compartment and $\dot{M}_{ij}$ is the flow rate between compartments i and j as discussed in Equation 3.6.A-4. The total enthalpy flow rate for the system is

$$\dot{H} = \sum_i \dot{H}_i \qquad \text{(Eq. 3.6.A-20)}$$

When energy transfer occurs between a compartment and the atmosphere, the relation used to calculate this flow is

$$\dot{H}_{atm,i} = \dot{M}_{ii} h^*_{ii} \qquad \text{(Eq. 3.6.A-21)}$$

Here $\dot{M}_{ii}$ represents the total flow from or to the atmosphere from compartment i and h^*_{ii} is the specific enthalpy of the upstream compartment (which may be either compartment i or the atmosphere depending upon the sign $\dot{M}_{ii}$). The total enthalpy flow rate to the atmosphere is

$$\dot{H}_{atm} \ \sum_i \ \dot{H}_{atm,i} \qquad \text{(Eq. 3.6.A-22)}$$

and the total amount of energy transferred to the atmosphere is

$$H_{atm}(t) = \int_o^t \dot{H}_{atm} \ dt \qquad \text{(Eq. 3.6.A-23)}$$

3.6.A.4.3 Compartment Heat Loads – Heat is generated within a compartment in the case where pumps or equipment are operating in that compartment. These heat loads are given with the input data as a constant heat rate (Btu/sec) for each compartment Q_{load}. These heat loads are assumed to be applicable throughout the problem under consideration.

3.6.A.4.4 Unit Coolers – Unit coolers or room coolers are present in many situations, especially in compartments that have equipment capable of generating large heat loads. Room coolers can have a variable start temperature which is specified in the input data. The coolers are usually set to begin operating when the compartment temperature exceeds some prescribed limit.

The cooling heat transfer rate is given by

$$\dot{Q}_{cool} = \alpha \ (T - T_{cool}) \qquad \text{(Eq. 3.6.A-24)}$$

Where T_{cool} is the cooler cold water inlet temperature, T is the temperature of the compartment, and α is the cooler constant (Btu/sec-°R). The cooler constant can be calculated from room cooler specifications and is assumed to be constant throughout the temperature ranges of the room atmosphere and the cooling water temperature.

3.6.A.4.5 Energy Balance – The energy balance given by the following equations is used to ensure that energy conservation is achieved.

$$E_{bal} = E_i + \dot{Q} dt - \dot{H}_{atm} dt - \dot{H}_B dt - E_i \ (0) \qquad \text{(Eq. 3.6.A-25)}$$

where E_I is the total energy in the ith compartment, E_i (0) is the initial compartment energy, and

$$\dot{Q} = \dot{Q}_c + \dot{Q}_{load} + \dot{Q}_{cool} \qquad \text{(Eq. 3.6.A-26)}$$

If an energy balance is achieved, then E_{bal} should be zero.

3.6.A.4.6 Blowout Panel Activation – Blowdown panels are treated as instantaneous one-way switches. Once a blowout panel set pressure is exceeded, the flowpath is open for the duration of the calculation. The actual activation of a blowout panel is made by setting the forward and reverse set pressures equal to zero once the forward set pressure has been exceeded.

3.6.A.4.7 Energy and Mass Conservation – Energy and mass conservation is then checked by calculating the following quantitites:

$$E_{bal} = \Sigma \ E_i + \int \dot{Q} \, dt + \int \dot{H}_{atm} dt - \int \dot{H}_B dt - E_{init} \qquad \text{(Eq. 3.6.A-27)}$$

$$M_{bal} = \Sigma \ M_i + \int \dot{M}_Q dt + \int \dot{M}_{atm} dt - \int \dot{M}_B dt - M_{init} \qquad \text{(Eq. 3.6.A-28)}$$

If all mass and energy transfer has been accounted for, the E_{bal} and M_{bal} should be zero (or a very small percentage of the total energy and mass due to computer round-off error.)

3.6.A.4.8 Eulerian Integration – The time-dependent quantities listed below are integrated according to the following general scheme:

$$X(T + \Delta t) = X(t) + \dot{X}(t)\Delta t \qquad \text{(Eq. 3.6.A-29)}$$

where X is any time dependent variable and $\dot{X}$ is its time rate of change. The variables integrated by FLUD are:

- $\dot{H}_B$ - blowdown enthalpy flow rate
- $\dot{M}_B$ - blowdown mass flow rate
- $\dot{E}$ - energy rate of change
- $\dot{H}_{atm}$ - atmospheric enthalpy flow rate
- $\dot{M}_a$ - air mass flow rate
- $\dot{Q}$ - Heat transfer rate
- $\dot{M}_v$ - vapor mass flow rate
- $\dot{M}_{atm}$ - atmospheric mass flow rate
- $\dot{M}_a$ - mass condensation rate

3.6.A.5 Thermodynamic Properties of Steam, Water, and Air.

FLUD uses steam, air, and water properties for various thermodynamic calculations which are performed during each step. The thermodynamic variables needed in FLUD calculations are:

- $e_a(T)$ - specific internal energy of air
- $P_{sat}(T)$ - saturation pressure of steam
- $v_{sat}(T)$ - saturation specific volume of steam
- $e_s(T,P)$ - specific internal energy of steam

$v_w(T)$ - specific volume of water

$e_w(T)$ - specific internal energy of water

$T_{sat}(P)$ - saturation temperature of steam

$T_{sat}(v)$ - saturation temperature of steam

$e_{sat}(T)$ - saturation specific internal energy of steam

$h_{sat}(T)$ - saturation specific enthalpy of steam

$h_{fg}(P)$ - enthalpy of vaporization of steam

The “unknown” quantities that can be used to calculate the above variables are the macroscopic compartment thermodynamic variables pressure, specific volume, and temperature, P, v, and T, respectively.

The air and water properties $e_a(T)$, $v_w(T)$, and $e_w(T)$ are calculated by fitting polynomials to data in the steam and gas tables (Refs. 3.6.A-2 and 3.6.A-3). The air property $e_a(T)$ was found to be adequately represented by a linear fit. This is no doubt due to the good “ideal gas” behavior of air. Thus,

$$e_a(T) = a_1 T \qquad \text{(Eq. 3.6.A-30)}$$

The water properties $v_w(T)$ and $e_w(T)$ and the steam properties $h_{sat}(T)$, $e_o(T)$, and $e_{sat}(T)$ are very nearly straight line functions, but small variations were accomdated by using third order spline polynomial fits of the general form:

$$\text{property}(T) = a_o + a_l T + a_2 T^2 + a_3 T^3 \qquad \text{(Eq. 3.6.A-31)}$$

For example, for $h_{fg}(P)$;

$$h_{fg}(P) = a_o + a_l P + a_2 P^2 + a_3 P^3 \qquad \text{(Eq. 3.6.A-33)}$$

The accuracy of the curve fits range between 0.01 percent and 4 percent for the various properties.

3.6.A.6 GOTHIC 4.0 HLP-001

GOTHIC 4.0 HLP-001 (Reference 3.6.A-8) is a state-of-the-art program that solves the conservation equations for mass, momentum and energy for multi-component, multi-phase flow. The phase balance equations are coupled by mechanistic models for interface mass, energy and momentum transfer that cover the entire flow regime from bubbly flow to film/drop flow, as well as single phase flows. The interface models allow for the possibility of thermal nonequilibrium between phases and unequal phase velocities. GOTHIC includes full treatment of the momentum transport terms in multi-dimensional models, with an optional one-dimensional turbulence model for turbulent shear and turbulent mass and energy diffusion. Conservation equations are solved for three fields:

- Steam/gas mixture
- Continuous liquid
- Liquid droplet

The program calculates the relative velocities between the separate but interacting fluid fields, including the effects of two-phase slip on pressure drop. The program also calculates heat transfer between phases, and between surfaces and the fluid. Liquid droplets are transported in the vapor/gas flow.

The three fluid fields may be in thermal nonequilibrium in the same computational cell. For example, saturated steam may exist in the presence of a superheated pool and sub-cooled drops. The code can model extremely dry noncondensable gases (down to steam partial pressures of 0.001 psia) and has a temperature range from –50 F^o to 8540 F^o.

The steam/gas mixture is referred to as the vapor phase and is comprised of steam and, optionally, up to eight different noncondensable gases including air and hydrogen. Mass balances are solved for each component of the steam/gas mixture, thereby providing the volume fraction of each type of gas in the mixture.

Solution of the equations is based upon nodalization of the region of interest, with the prinicpal element of a model being a computational volume. The program features a flexible noding scheme that allows computational volumes to be treated as lumped parameter, one-, two- or three-dimensional, or any combination of these within a single model. As a minimum, a GOTHIC model consists of at least one lumped parameter volume. Subdivision of a volume into a one-, two- or three-dimensional mesh is based on orthogonal coordinates. Adjacent cells in a subdivided volume communicate through parameters defined by discretization of the governing equations. Separate volumes communicate through what are referred to as junctions or flow paths. A separate set of momentum equations are solved for junctions.

GOTHIC has been verified against the applicable portions of ANSI/ANS 56.10-1982 (Reference 3.6.A-10) for subcompartment pressurization analysis.

3.6.A.7 STEM TRAVEL

STEM_TRAVEL (Reference 3.6.A-9) was developed by HL&P (historical context) to facilitate the use of GOTHIC in subcompartment pressure/temperature (P/T) transient analysis. The code calculates the panel height corresponding to a given subcompartment pressure profile generated from the GOTHIC computer code. The vent paths are modeled by an equivalent panel height, and in turn, translated into an equivalent stem travel. GOTHIC generates a new pressure profile based on the new stem travel profile. Iterations between the two codes are done manually until convergence between the stem travel and pressure profiles is obtained. The panel height is obtained from the governing equation for dynamic vent paths as discussed in Appendix E of ANSI/ANS 56.10-1982, "Subcompartment Pressure and Temperature Transient Analysis in Light Water Reactors," Reference 3.6.A-10.

$$M\frac{d^2h}{dt^2} = A_o(P_{in} - P_{out} - P_m + \frac{W^2}{\rho A_o^2})$$

where:

M	=	mass of panels
h	=	current vertical panel displacement,
A_o	=	fully open vent area,
P_m	=	weight per unit area of the panels,
P_{in}	=	static pressure at current time in the region beneath the panel,
t	=	time,
P_{out}	=	static pressure in the region above the panel,
W	=	mass flow rate, and
ρ	=	fluid mass density.

The equation is simplified to the following form in STEM_TRAVEL:

$$h = h_o + v_o\tau + \frac{g_c}{m}(\frac{1}{6}\frac{P_{in} - P_{in,0}}{\tau}\tau^3 + \frac{P_{in,o} - P_{out} - P_m}{2}\tau^2)$$

where

h	=	current vertical panel displacement,
h_o	=	panel displacement at beginning of time step,
v_o	=	panel velocity at beginning of time step,
τ	=	time step size, Δt,
g_c	=	gravitational constant,
m	=	mass per unit area of the panels,
P_m	=	weight per unit area of the panels,
P_{in}	=	static pressure at current time in the region beneath the panel,
$P_{in,0}$	=	static pressure below the panel at beginning of time step, and
P_{out}	=	static pressure in the region above the panel.

REFERENCES

Appendix 3.6.A:

3.6.A-1 Reif, F.J. Fundamentals of Statistical and Thermal Physics, McGraw-Hill Book Co., p. 183.

3.6.A-2 Kennan, J.H. et al, Steam Tables, John Wiley & Sons, Inc., New York, 1969.

3.6.A-3 Keepan, J.H., and J. Kaye, Gas Tables, John Wiley & Sons, Inc., New York, 1948.

3.6.A-4 Bechtel Topical Report BN-TOP-4 Rev. 1, October 1977, "Subcompartment Pressure and Temperature Transient Analysis". This report was approved by the NRC in February, 1979.

3.6.A-5 Burnett, T.W.T., et. al., "LOFTRAN Code Description", WCAP-7907-P-A (Proprietary Class 2), WCAP-7907-A, (Proprietary Class 3), April 1984.

3.6.A-6 Bilanin, W.J., "General Electric Mark III Pressure Suppression Containment System Analytical Mode", General Electric Topical Report NEDO 20533, June 1974.

3.6.A-7 Sharma, D.F., "Technical Description – Annulus Pressurization Load Adequacy Evaluation", General Electric Topical Report NEDO 24548, January 1979.

3.6.A-8 George, T.L., et al, "GOTHIC 4.0 HLP-001, Containment Analysis Package, "Developed by Numerical Application, Inc. for the Electric Power Research Institute, September, 1993.

3.6.A-9 Futschik, M.W., "STEM_TRAVEL", developed by Houston Lighting & Power Company, February, 1995.

3.6.A-10 ANSI 56.10-1982, "Subcompartment Pressure and Temperature Transient Analysis in Light Water Reactors."

TABLE 3.6.A-1

MAIN STEAM LINE BREAK

SHORT-TERM MASS AND ENERGY RELEASE RATES IN IVC

Time (sec)	Mass Flowrate (lbm/sec)	Enthalpy (Btu/lbm)
0.0	0.0	1181.80
0.0052	10446.95	1178.68
0.0055	9358.21	1169.23
0.006	7849.90	1154.29
0.0065	6782.15	1143.06
0.007	6144.52	1138.21
0.107	6414.60	1165.94
0.207	6494.62	1167.96
0.407	6524.78	1173.13
0.607	6408.85	1176.33
0.807	11417.89	706.47
0.907	11810.01	697.57
1.0	11759.46	693.90
2.0	12928.62	679.37
3.0	13447.40	658.52
4.0	14025.18	644.32
5.0	14540.07	636.16
6.0	14976.29	630.52
7.0	15360.99	628.91
8.0	15365.68	626.61
9.0	15271.03	624.75
10.0	15454.93	620.97
12.0	15798.19	616.86
14.0	15976.33	614.50
16.0	16123.07	613.25
18.0	16077.40	612.73
20.0	16197.69	610.95

CN-2822

TABLE 3.6.A-2

IVC SUBCOMPARTMENT NODAL DESCRIPTION

FOR MAIN STEAM LINE BREAK

Volume Number	Volume ft^3	Initial Conditions Temp. °F	Pressure psia	Humidity Percent	Flow Path	Flow Area ft^2	Flow Coeffi-cient	L/A ft^{-1}	Calculated Peak Press. psia	Break Type
1.	3588.5	104.0	14.7	20					24.6	MSLB
2.	1977.5	104.0	14.7	20	2 1	75.45	0.79	0.05	24.6	MSLB
3.	5530.95	104.0	14.7	20	3 2	54.16	0.83	0.18	24.6	MSLB
4.	2558.5	104.0	14.7	20	4 3	210.37	0.81	0.024	24.7	MSLB
					4 5	50.32	0.90	0.26		
					4 6	256.84	0.82	0.02		
5.	1453.0	104.0	14.7	20	5 3	115.72	0.81	0.035	25.0	MSLB
					5 7	115.42	0.81	0.04		
6.	2221.7	104.0	14.7	20	6 7	35.47	0.88	0.28	24.8	MSLB
					6 8	195.42	0.80	0.05		
7.	1262.26	104.0	14.7	20	7 9	90.29	0.79	0.09	27.7	MSLB
8.	8131.36	104.0	14.7	20	8 9	101.59	0.865	0.077	21.0	MSLB
					8 3P	172.5	0.65	0.038		
9.	5606.85	104.0	14.7	20	9 2P	178.	0.72	0.04	20.2	MSLB
					9 1P	15.5	0.54	0.48		
3P.	N/A	104.0	14.7	N/A	)					
2P.	N/A	104.0	14.7	N/A	)	PRESSURE BOUNDARY CONDITION				
1P.	N/A	104.0	14.7	N/A	)					

TABLE 3.6.A-3

NODE 10. VARIABLE NODE PARAMETERS

Variable Height of the Panel (ft)	Variable Vent Area (ft2)	Variable Volume (ft3)
0.0	0.0	173.92
0.4	5.2	298.71
0.8	10.4	423.50
1.05	14.32	501.50
1.30	19.59	579.50
1.55	26.88	657.49
1.92	42.61	772.92
2.92	100.98	1084.90
3.92	161.97	1396.88
5.00	227.84	1733.82

TABLE 3.6.A-4

NODE 11, VARIABLE NODE PARAMETERS

Variable Height of the Panel (ft)	Variable Vent Area (ft2)	Variable Volume (ft3)
0.0	0.0	158.38
0.4	5.2	275.37
0.8	10.4	392.37
1.5	19.5	597.11
1.92	24.96	719.95
2.32	33.48	836.95
2.72	48.66	953.94
3.00	63.25	1035.84
4.00	121.24	1328.32
5.00	179.24	1620.81

TABLE 3.6.A-5

MASS/ENERGY RELEASE RESULTS FOR A 4.05 FT2

SPLIT STEAM LINE BREAK AT 0% POWER

Time	Mass Flow Rate (lbm/sec)	Enthalpy (Btu/lbm)
0.0	10143.6	1184.98
2.5	9336.6	1189.68
5.0	8550.8	1192.87
7.5	7906.5	1195.46
10.0	2506.3	1197.98
15.0	2084.8	1201.55
20.0	1780.7	1203.46
25.0	1561.5	1204.42
30.0	1397.9	1204.74
35.0	1270.5	1204.80
40.0	1170.3	1204.73
45.0	1091.7	1204.36
50.0	1030.3	1204.16
60.0	942.5	1203.60
80.0	855.7	1202.88
100.0	821.5	1202.43
150.0	786.8	1202.08
200.0	743.6	1201.45
250.0	696.9	1200.88
300.0	628.2	1199.62
310.0	595.1	1198.62
350.0	595.1	1198.62
351.0	141.3	1198.86
1800.0	141.3	1198.86
2802.0	0.0	0.0

TABLE 3.6.A-6

IVC COMMON CORRIDOR AREA SUBCOMPARTMENT ANALYSIS

	Initial Conditions (in volumes)					Flow Parameters			
Volume Number	Volume (ft^3)	Temp (°F)	Pressure (psia)	Relative Humidity (percent)	**Peak Pressure (psia)**	Flow Path	Area (ft^2)	Flow Coefficient	Inertia Coefficient (L/A) (ft^{-1})
1	4950	129	14.7	50	19.2	1-2	41.1	0.83	0.16
2	4950	129	14.7	50	18.5	2-3	41.1	0.83	0.16
3	4950	129	14.7	50	17.9	3-4	41.1	0.83	0.16
4	4950	129	14.7	50	17.4	4-5	54.0	0.83	0.73
5	7024	129	14.7	50	17.1	5-6	18.9	0.82	0.63
6	1.0E22	129	14.7	50	N/A (atmosphere)				

APPENDIX 3.6.B

HIGH ENERGY LINE BREAK EFFECTS

3.6.B.1 Isolation Valve Cubicle (IVC).

Break Locations

A. Breaks are conservatively postulated in the Main Steam (MS) and Feedwater (FW) System branch piping at terminal ends and each intermediate fitting (e.g., short-and long-radius elbows, tees and reducers, welded attachments and valves).

B. Breaks were postulated in accordance with Section 3.6.2.1.1(2) criteria for the Steam Generator (SG) Blowdown System piping (Figure 3.6.1-1 Sheets 6A through 6D). Since the auxiliary feedwater (AFW) pumps are not used for normal plant operational modes, the criteria of Section 3.6.1.1(1) was used to determine the high energy piping and postulated break locations between the isolation check valve and the containment penetration, as shown on Figures 3.6.1-1 Sheets 4A through 4D.

Effect Analysis

A. Pipe Whip/Jet

An evaluation was performed to identify those systems, structures and components necessary for safe shutdown following the jet and whip effect of the breaks postulated above.

Due to the complete separation design concept of the IVC structure and the multiple (4) train systems (AFW, MS, Main FW, SG blowdown) enclosed by the structure, all mechanical equipment (piping, pumps heating, ventilation, and air conditioning (HVAC), etc.), control (main steam isolation valves [MSIVs], main feedwater isolation valves [MFIVs], containment isolation valves) and electrical (power and control circuits) devices within an affected compartment do not require additional protection for the direct jet or whip interaction from the postulated break locations. However, in order to prevent cross communication between cubicles and to maintain the complete separation concept, the IVC walls, slabs and floors were analyzed to withstand the direct pipe whip and jet effects. Therefore, no additional protective devices are necessary.

B. Flooding

A review of the high energy lines within the IVC showed that a non-mechanistic break of the main FW line in each cubicle determined the maximum flood level in that cubicle. Blowdown was conservatively calculated from both SG and the FW pumps as well as consideration of AFW flow into the SG subsequent to a low level signal in the affected SG.

In the AFW compartments the maximum flood level calculated is 28 ft above the cubicle floor. Although the affected train of AFW could be damaged and a second AFW could be inoperable due to a limiting single failure, the remaining AFW would be sufficient for safe shutdown following the postulated main FW line break. The maximum calculated flood level for the North stairwell (or common) compartment of the IVC following an main FW line break is 1.9 ft above the floor. Since the penetration openings between the pump rooms and the North stairwell are designed to be watertight, this flood level does not affect essential systems and components within the IVC.

Therefore, safe shutdown is assured following the flooding effects from postulated high energy line breaks within the IVC.

C. Pressure/Temperature Effects

See Appendix 3.6.A for a discussion of the pressure and temperature parameters for postulated high energy line breaks within the IVC. These parameters are used for structural design and environmental qualification of enclosed safety-related equipment. (See Section 3.11 for the environmental qualification of safety-related equipment.)

3.6.B.2 MECHANICAL AUXILIARY BUILDING (MAB).

Break Locations

A. Breaks are conservatively postulated in the non-nuclear auxiliary steam (AS) piping at terminal ends and each intermediate fitting (e.g., short-and long-radius elbows, tees and reducers, welded attachments and valves).

B. In accordance with the criteria described in Section 3.6.2.1.1(2), breaks were postulated in Chemical Volume and Control System (CVCS) letdown line located within the MAB (Figure 3.6.1-1 Sheet 9B). In addition, breaks were initially postulated in the CVCS centrifugal charging pump (CCP) discharge piping per Section 3.6.2.1.1.(2).

Effects Analysis

A. Pipe Whip/Jet

Safety-related system, components and structures impacted by the jet and whip from the above postulated letdown line and AS line breaks were either analyzed to withstand the jet/whip effects (e.g., impacted protective walls and slabs) or determined not essential for each postulated break (e.g., safe shutdown could be obtained with loss of the impacted safety-related components). Subsequent to the initial postulated break locations in the CVCS CCP discharge piping, and evaluation demonstrated an insufficient level of stored energy exists to impair the safety function of any structure, system or component to an unacceptable level.

B. Flooding

High energy lines located inside the MAB are limited to auxiliary steam, liquid waste processing and CVCS piping. A break in these lines will be detected by redundant temperature elements when a rise in area temperature occurs following the postulated event. Break flow will be terminated by automatic isolation valves on these lines (see Section 9.3.4.1.3.5 and 9.5.9.3 for additional details). No essential components required for safe shutdown will be flooded as a result of an AS or liquid waste processing line break. In the case of a CVCS line break, damage as a result of flooding will be limited to equipment located in the vicinity of the break. The redundant CVCS train would be available for safe shutdown. An evaluation has been performed that demonstrates that even if all of CVCS is assumed to be lost as a result of the event, safe shutdown can still be achieved.

Flooding of both reactor makeup water pumps as a result of the failure of nonseismic Category I pipe in the reactor makeup water tank compartment has also been analyzed. Details are provided in Section 9.1.3.3.2.

C. Pressure/Temperature

Subcompartment pressure and temperature analysis has been performed for the high energy breaks postulated for the CVCS letdown and the AS piping using conservative non-mechanistic or “break everywhere” criteria. The methodology used is similar to the methodology used in the IVC subcompartment evaluation described in Appendix 3.6.A. The analysis for the MAB took credit for the safety-related high temperature detectors and associated isolation valve interlocks in the affected areas that limit the mass and energy release.

Pressure and temperature profiles for the letdown heat exchanger (HX) room are presented in Figures 3.6.2-5 and 3.6.2-6.

The result of the subcompartment analysis is used as the basis for the environmental qualification of electrical equipment (Section 3.11) as well as factored into the design of affected structures.

3.6.B.3 Reactor Containment Building (RCB).

Break Location

A. Partial stress summaries and break types for the Containment High Energy Piping Systems are presented in Table 3.6.2-1.

B. Break locations and types are shown for the Containment High Energy Piping Systems in Figure 3.6.1-1.

Effects Analysis

A. Pipe Whip/Jet

Safety-related systems, structures and components impacted by the whip/jet from the above postulated breaks are analyzed to withstand the whip/jet effects (e.g., impacted protective walls and slabs), determined not to be essential for each postulated break (e.g., safe shutdown could be obtained with loss of the impacted safety-related components) or determined to be essential and the appropriate protective devices (pipe whip restraint, jet barrier, etc.) incorporated into the plant design.

The pipe whip restraints and other necessary devices (e.g., jet barriers) that have been incorporated into the South Texas Project Electric Generating Station (STPEGS) design are shown in Figures 3.6.1-1 along with the applicable break location and restraint load summary.

B. Flooding

The Containment flooding analysis has shown that the maximum volume of water discharged to the RCB occurs as a result of a Loss of Coolant Accident (LOCA), and water from the Reactor Coolant System (RCS), the refueling water storage tank and the accumulators is assumed to spill onto the RCB floor. This analysis has been performed in accordance with the criteria and methodology described in Sections 3.4.3 and 3.4.4.

An evaluation has been performed which confirms the ability to safely shutdown the plant taking into account the equipment which is expected to be flooded as a result of the LOCA. This evaluation is presented in Table 3.6.B-1.

C. Pressure/Temperature Effects

Section 6.2.1 describes the pressure and temperature effects for selected compartments inside the Containment.

TABLE 3.6.B-1

EQUIPMENT BELOW RCB MAXIMUM FLOOD ELEVATION

Equipment Identification	Service Description	Class 1E Powered	P&ID No.	Elev.	Safety Significance of Flooding
B1CC-MOV-0139	Cooling Water Inlet Valve to RCFC 12B	Yes	9F05018	-9′-6″	None – see Note 2.*
B1CC-MOV-0142	Cooling Water Discharge Valve from RCFC 12B	Yes	9F05018	-8′-0″	None – see Note 2.*
B1CC-MOV-0143	Cooling Water Inlet Valve to RCFC 11B	Yes	9F05018	-9′-6″	None – see Note 2.*
B1CC-MOV-0146	Cooling Water Discharge Valve from RCFC 11B	Yes	9F05018	-9′-6″	None – see Note 2.*
N1CC-FE-4553	Cooling Water Discharge from RCFC 11B	No	9F05018	-9′-6″	None – flooding has no effect on component.
N1CC-TE-4554	Cooling Water Discharge from RCFC 11B	No	9F05018	-9′-6″	None – component not required post-accident.
N1CC-FE-4555	Cooling Water Discharge from RCFC 12B	No	9F05018	-8′-0″	None – flooding has no effect on component.
N1CC-TE-4556	Cooling Water Discharge from RCFC 12B	No	9F05018	-8′-0″	None – component not required post-accident
N1CC-TE-4642	Reactor Coolant Drain Tank HX CCW Outlet	No	9F05021	-4′-9″	None – component not required post-accident
N1CC-FE-4643	Reactor Coolant Drain Tank HX CCW Outlet	No	9F05021	-4′-9″	None – flooding has no effect on component
N1CC-FI-4643	Reactor Coolant Drain Tank HX CCW Outlet	No	9F05021	-6′-9″	None – component not required post-accident

* See notes at end of table.

TABLE 3.6.B-1 (Continued)

EQUIPMENT BELOW RCB MAXIMUM FLOOD ELEVATION

Equipment Identification	Service Description	Class 1E Powered	P&ID No.	Elev.	Safety Significance of Flooding
N1CV-FIS-0166	RCP 1A No. 2 Seal Leakoff Flow	No	9F05005	-6′-9″	None – component not required post-accident.
N1CV-FIS-0167	RCP 1B No. 2 Seal Leakoff Flow	No	9F05005	-8′-7″	None – component not required post-accident.
N1CV-FIS-0168	RCP 1C No. 2 Seal Leakoff Flow	No	9F05005	-8′-7″	None – component not required post-accident.
N1CV-FIS-0169	RCP 1D No. 2 Seal Leakoff Flow	No	9F05005	-6′-9″	None – component not required post-accident.
N1ED-LT-7811	Containment Secondary Sump Level	No	9F05030	-6′-9″	None – component not required post-accident.
N1ED-LT-7812	Containment Normal Sump Level	No	9F05030	-6′-9″	None – component not required post-accident.
A1ED-LE-7839	Containment Water Level (NR) Containment Normal Sump	Yes	9F05030	Note 3	None – when component becomes flooded, its function in detection of a small break LOCA is no longer necessary.
C1ED-LE-7840	Containment Water Level (NR) Containment Secondary Sump	Yes	9F05030	Note 3	None – when component becomes flooded, its function in detection of a small break LOCA is no longer necessary.
N1HC-TE-9629	Reactor Cavity Vent Fan Cooling Coil VHX004 Discharge	No	9V00022	-6′-0″	None – component not required post-accident.
N1HC-TE-9631	Reactor Cavity Vent Fan Cooling Coil VHX003 Discharge	No	9V00022	-6′-0″	None – component not required post-accident.
N1RA-RE-8053	Area Radiation Monitor	No	None	-6′-9″	None – component not required post-accident.
N1RA-RI-8053	Area Radiation Monitor	No	None	-6′-9″	None – component not required post-accident.
N1RC-PT-0669	Pressurizer Relief Tank Pressure	No	9F05004	-6′-9″	None – component not required post-accident.

TABLE 3.6.B-1 (Continued)

EQUIPMENT BELOW RCB MAXIMUM FLOOD ELEVATION

Equipment Identification	Service Description	Class 1E Powered	P&ID No.	Elev.	Safety Significance of Flooding
N1RC-PL-0670	Pressurizer Relief Tank Level	No	9F05004	-6′-9″	None – component not required post-accident.
A1SI-MOV-0039A	Accumulator 1A Discharge Isolation Valve	Yes	9F05016	-7′-7″	None – see Note 2.
B1SI-MOV-0039B	Accumulator 1B Discharge Isolation Valve	Yes	9F05016	-7′-7″	None – see Note 2.
C1SI-MOV-0039C	Accumulator 1C Discharge Isolation Valve	Yes	9F05016	-7′-7″	None – see Note 2.
A1SI-LE-3925	Containment Water Level (WR) at Emergency Sump 1A	Yes	9F05013	Note 4	None – only parts designed for level detection are below flood level.
B1SI-LE-3926	Containment Water Level (WR) at Emergency Sump 1B	Yes	9F05014	Note 4	None – only parts designed for level detection are below flood level.
C1SI-LE-3927	Containment Water Level (WR) at Emergency Sump 1C	Yes	9F05014	Note 4	None – only parts designed for level detection are below flood level.
N1WL-PSH-4900	Reactor Coolant Drain Tank Pressure	No	9F05022	-6′-9″	None – component not required post-accident.
N1WL-PT-4900	Reactor Coolant Drain Tank Pressure	No	9F05022	-6′-9″	None – component not required post-accident.
N1WL-LT-4901	Reactor coolant Drain Tank Level	No	9F05022	-6′-9″	None – component not required post-accident.
N1WL-PT-4904	Reactor Coolant Drain Tank Pumps Discharge	No	9F05022	-6′-9″	None – component not required post-accident.
N1WL-FE-4905	Reactor Coolant Drain Tank Pumps Discharge	No	9F05022	-5′-6″	None – flooding has no effect on component.
N1WL-FT-4905	Reactor Coolant Drain Tank Pumps Discharge	No	9F05022	-6′-9″	None – component not required post-accident.
N1WL-TE-4906	Reactor Coolant Drain Tank HX Inlet	No	9F05022	-5′-6″	None – component not required post-accident.

TABLE 3.6.B-1 (Continued)

EQUIPMENT BELOW RCB MAXIMUM FLOOD ELEVATION

Equipment Identification	Service Description	Class 1E Powered	P&ID No.	Elev.	Safety Significance of Flooding
N1WL-TE-4906A	Reactor Coolant Drain Tank HX Outlet	No	9F05022	-10′-3″	None – component not required post-accident.
N1WL-FV-4907	Reactor Coolant Drain Tank Discharge Valve	No	9F05022	-10′-3″	None – component not required post-accident.
N1WL-FY-4907	Reactor Coolant Drain Tank Discharge Valve Solenoid	No	9F05022	-10′-3″	None – component not required post-accident.
N1WL-FV-4910	Reactor Coolant Drain Tank Recirculation Valve	No	9F05022	-10′-3″	None – component not required post-accident.
N1WL-FY-4910	Reactor Coolant Drain Tank Recirculation Valve Solenoid	No	9F05022	-10′-3″	None – component not required post-accident.
N1WL-LV-4911	Reactor Coolant Drain Tank Level Control Valve	No	9F05022	-10′-3″	None – component not required post-accident.
N1WL-LY-4911	Reactor Coolant Drain Tank Level Control Valve Solenoid	No	9F05022	-10′-3″	None – component not required post-accident.

1) Abbreviations used are as follows:

CCW	Component Cooling Water	NR	Narrow Range
HHSI	High Head Safety Injection	RCB	Reactor Containment Building
HX	Heat Exchanger	RCFC	Reactor Containment Fan Cooler
LHSI	Low Head Safety Injection	RCP	Reactor Coolant Pump
LOCA	Loss of Coolant Accident	RHR	Residual Heat Removal
		WR	Wide Range

2) Valve is normally open (position required for accident) with power removed by administrative control. Spurious closure is not credible. Valve power lockout control and valve control are provided in main control room.

3) Level instrument installed vertically; top is below flood level.

4) Level instrument installed vertically; top is above flood level.

5) Maximum flood elevation is El. -4′8″.

3.7 SEISMIC DESIGN

3.7.1 Seismic Input

3.7.1.1 Design Response Spectra. Two earthquake motions have been considered, namely, Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE). Definitions of these earthquake motions are given in Section 2.5, Glossary.

The peak accelerations associated with SSE and OBE have been established based on the seismicity evaluation described in Section 2.5. The expected peak horizontal acceleration at this site is less than 0.10g. The peak horizontal accelerations of 0.10g and 0.05g incorporated in the design response spectra for the SSE and OBE, respectively, comply with Appendix A, "Reactor Site Criteria," to 10CFR100. The ground acceleration as represented by the spectral acceleration at 33 Hz is 0.1g for both the horizontal and the vertical directions. At 50 Hz the vertical spectral acceleration is reduced to two-thirds of the horizontal acceleration.

Horizontal design response spectra for 1-percent, 2-percent, 4-percent, 7-percent, and 10-percent spectral damping values are presented on Figures 3.7-1 and 3.7-2 for the SSE and OBE, respectively. Vertical design response spectra for the SSE and OBE for the same damping values are presented on Figures 3.7-3 and 3.7-4. The design response spectra are developed in accordance with Regulatory Guide (RG) 1.60, Revision 1.

3.7.1.2 Design Time-History. Artificial accelerograms, whose spectra essentially envelop the horizontal and vertical design response spectra presented in Section 3.7.1.1, have been generated from actual earthquake acceleration motions by means of selective amplification and phasing of their Fourier components. The time-histories are discretized at a time-step of 0.005 second and have a duration of 10 seconds. Figure 3.7-5 shows the horizontal and vertical SSE artificial time histories. The peak vertical acceleration is two-thirds of the peak horizontal acceleration. For the OBE, the ordinates of the figure need to be multiplied by a factor of 0.5.

The horizontal and vertical SSE response spectra of the artificial accelerograms presented on Figures 3.7-6 through 3.7-10 and on Figures 3.7-11 through 3.7-15 respectively were calculated at 242 points in a frequency range of 0.5 to 50 Hz as indicated below.

Frequency Range	No. of Points
0.5 - 2.5	80
2.5 - 8.0	70
8.0 - 25.0	70
25.0 - 35.0	12

35.0 - 50.0	10

Comparisons of the calculated spectra for the artificial accelerograms with the corresponding design spectra for damping ratios of 0.01, 0.02, 0.04, 0.07, and 0.10 are presented in Figures 3.7-6 through 3.7-15, respectively. None of the points are lower than the prescribed allowable of 10 percent below the design response spectrum, and generally the number of points that are lower than the design response spectrum is less than the prescribed maximum of five points. However, there are two minor exceptions. The first exception is that for the vertical response spectrum for 4 percent damping, six points rather than five are lower than the design spectrum by 3 to 4 percent at very low frequencies. This slight departure is inconsequential, and the calculated spectra is considered to be adequately matched to the design spectra. The second exception is that the computed spectra are not defined in the very low frequency range of less than 0.4 Hz. Since such low frequencies could not be incorporated into the relatively short duration (10 seconds) of the artificial accelerograms. However, this range of undefined spectral response is of no concern since it is confined to very low frequencies which are insignificant with respect to design.

For soil/structure interaction analyses (described in Section 3.7.2.4), acceleration time-histories were obtained at the base of the idealized soil profile by a deconvolution process. These time-histories were used as input for finite element analyses of the idealized soil profile from which response spectra were calculated from these analyses at finished grade and at foundation levels in the free-field. Comparisons of the calculated OBE horizontal response spectra at finished grade for the three sets of soil properties with the design response spectrum are given on Figure 3.7-15A and 3.7-15B. Comparisons of the envelope of the calculated response spectra at foundation levels in the free-field with 60 percent of the design response spectrum are given on Figure 3.7-15C. These comparisons, which are for the idealized soil profile, show compliance with Nuclear Regulatory Commission (NRC) criteria. Compliance with the criteria is also obtained for response spectra calculated in the free-field of the soil/structure interaction system.

3.7.1.2.1 Bases for Site Dependent Analysis: A site dependent analysis was not used to evaluate either the level of ground surface acceleration or to develop the design response spectra.

A site dependent analysis was used to evaluate the liquefaction potential; a summary of this analysis is provided in Section 2.5.

3.7.1.3 Critical Damping Values. The percentages of critical damping values applicable for structural components and systems of Category I structures are those listed in RG 1.61, October 1973, and included as Table 3.7-1.

Damping values used for the seismic analysis of the Nuclear Steam Supply System (NSSS) equipment are listed in Table 3.7-7. These are consistent with the damping values recommended in RG 1.61 except in the case of the primary coolant loop system components and large piping (excluding reactor pressure vessel [RPV] internals), for which the damping value of 4 percent for the SSE is used as established in testing programs reported in Reference 3.7.1-2. The damping values for control rod drive mechanisms (CRDM) and the fuel assemblies of the NSSS used in the Reactor Coolant System (RCS) analysis are given in Section 3.7.3B.15

Tests on fuel assembly bundles justified conservative component damping values of 7 percent for OBE and 10 percent for SSE used in the fuel assembly component qualification. Documentation of the fuel assembly tests is given in Reference 3.7.3-7.

Damping characteristics for the soil at this site are determined by laboratory tests on representative samples of applicable soil strata (refer to Section 2.5.4).

3.7.1.4 Supporting Media for Seismic Category I Structures. Soil conditions at the South Texas Project Electric Generating Station (STPEGS) site are described in Section 2.5.1.2.2. Generally, the supporting media consist of alternating layers of stiff to hard clays and dense silts and sand which extend to depths of several thousand feet. Three soil profiles showing various layers are presented in Figures 2.5.1-3 through 2.5.1-5. The locations of these profiles are given on Figure 2.5.1-2.

3.7.1.4.1 Category I Structures: The Category I structures that are soil-supported are listed below for each unit.

1. Reactor Containment Building (RCB)

2. Mechanical-Electrical Auxiliaries Building (MEAB)

3. Fuel Handling Building (FHB)

4. Diesel Generator Building (DGB)

5. Essential Cooling Water (ECW) Intake and Discharge Structures

6. Auxiliary Feedwater Storage Tank (AFST)

7. Class lE Underground Electrical Raceway System

8. Category I Underground Piping System

3.7.1.4.2 Foundation Embedment, Dimension of Foundation, and Total Height of Structures:

Building	Embedment Depth (ft)	Maximum* Height (ft)	Base Dimension
(1) RCB	59.25	262.25	166 ft dia x 18 ft thick mat

(2) MEAB	24	92	320 ft x 252 ft x 6 ft thick basemat (appx)
(3) FHB	63.75 max 14.00 min	156.75	184 ft x 87 ft x 6.75 ft thick basemat

* Includes basemat thickness

Building	Embedment Depth (ft)	Maximum* Height (ft)	Base Dimension
(4) DGB	8	87	82 ft x 107 ft x 5 ft thick basemat (appx)
(5a) ECW Intake Structure	28 max 11 min	49	136 ft x 79.5 ft x 4 ft thick mat
(5b) ECW Discharge Structure	10 (appx) max 4 (appx) min	23 (appx)	33 ft x 50 ft (appx) x 2 ft to 4 ft thick mat
(6) AFST	9	54	61 ft-0 in. dia (appx) x 4 ft thick

* Includes basemat thickness

3.7.1.4.3 Soil Properties - Shear-Wave Velocity, Shear Modulus, and Density: Shear-wave velocities and shear moduli for various layers of soil are discussed in Section 2.5.4.7 and presented in Table 2.5.4-27, and densities are presented in Table 2.5.4-1. It is noted that for the seismic analysis of buried structures the pertinent shear wave velocity is the velocity corresponding to depths of 400 to 500 ft below the ground surface (Ref. 3.7.1-3). The shear wave velocity at this depth range was not part of the site specific soil evaluation. Therefore, the conservative lower bound of 2000 ft/sec was used in such analyses.

3.7.2 Seismic System Analysis

3.7.2.1 Seismic Analysis Methods. The seismic analyses of all Category I structures identified in Section 3.7.1.4.1 have been performed using either the modal time-history or the response spectrum method, as discussed below. The bases for the seismic analyses of the RCB, MEAB, FHB, and DGB, collectively designated as the power block structures, are time history analyses based on the foundation motions developed from the finite element method for soil/structure interaction (SSI) analyses discussed in Section 3.7.2.4. The seismic analysis of the Essential Cooling Water Intake Structure (ECWIS), is a modal time-history analysis based on the elastic half-space method for SSI. The free-field surface ground motions, increased by 20 percent, are used as input at the foundation basemat of a lumped-parameter model of the ECWIS. The seismic analyses of the ECW Discharge Structure is based on the equivalent static method utilizing 1.5 times the peak spectral accelerations from the RG 1.60 spectra. The seismic analysis of the AFST is based on modal response spectrum analysis utilizing the free-field design spectra to define input at the fixed-base of a lumped-parameter structural model. The methods used for seismic analysis of Category I structures are summarized in Table 3.7-2. Seismic analysis of components and equipment provided by the Nuclear Steam Supply System vendor are discussed in Section 3.7.3.

The seismic response within the power block structures is determined by using the motions calculated at the base of the foundations from the finite element SSI analyses (first-step analysis) as input to the detailed three-dimensional lumped-parameter mathematical model of each building (second-step analysis). Within each structure, time-history acceleration records are obtained from the second-step analysis at all major floor levels and other locations necessary for the seismic analyses of systems and components. Response spectra are calculated for each of these time-history records for subsequent use in modal response spectrum analyses for subsystems.

A sufficient number of nodal points and degrees of freedom have been taken into consideration to define the motion within each structural model. In all cases, either the number of degrees of freedom has been chosen more than twice the number of modes with frequencies less than 33 Hz, or the inclusion of additional modes will not result in more than a 10-percent increase in responses.

The detailed seismic analyses of each structure include the effects of rocking at the base of the structures. Rotational (rocking) motion together with translational motion as determined from the first step SSI analyses, are input to the detailed mathematical models of each structure. Hence, the seismic analyses include rocking input directly, and all floor-level horizontal acceleration responses inherently include the translational effects of rocking.

Torsional effects have been taken into account by the torsional response obtained through the torsional degrees of freedom incorporated at the base of and within the three-dimensional lumped-parameter models. This is discussed further in Section 3.7.2.3.

Seismic analyses of structures have been performed using the computer program STRUDL DYNAL (Ref. 3.7.2-1).

3.7.2.1.1 Time-History Method: As described in Section 3.7.2.3 for seismic analysis of superstructures, three-dimensional, lumped-parameter models have been used. For such structures, the equations of dynamic equilibrium can be expressed in matrix form as:

$$[M]\ \{\ddot{x}\} + [C]\ \{\dot{x}\} + [K]\ x = -\ [M]\ [T]\ \{\ddot{U}_g\} \qquad \text{(Eq. 3.7.2-1)}$$

where:

[M] = Mass matrix

[C] = Damping matrix

[K] = Stiffness matrix

{x} = Vector of displacements relative to the ground

$\{\ddot{U}_g\}$ = Vector of ground translational and rotational accelerations

[T] = Transformation matrix. For the case of two-step time history analyses where the base motion consists of translational and rotational accelerations, [T] has the following form:

$$[T] = \begin{bmatrix} [T_1] \\ [T_2] \\ \cdot \\ \cdot \\ \cdot \\ [T_n] \end{bmatrix}$$

$$[T_i] = \begin{bmatrix} 1 & 0 & 0 & 0 & \Delta Z & -\Delta Y \\ 0 & 1 & 0 & -\Delta Z & 0 & \Delta X \\ 0 & 0 & 1 & \Delta Y & -\Delta X & 0 \\ 0 & 0 & 0 & 1 & 0 & 0 \\ 0 & 0 & 0 & 0 & 1 & 0 \\ 0 & 0 & 0 & 0 & 0 & 1 \end{bmatrix}$$

Where ΔX, ΔY, and ΔZ are coordinate differences between the ith node and the point of rigid-body rotation at the base.

The following transformation is defined:

$$\{x\} = [\phi]\ \{q\} \qquad \text{(Eq. 3.7.2-2)}$$

where:

$[\phi]$ = Mode shape matrix

{q} = Vector of generalized coordinates

The mode shape matrix is defined to have the following properties:

$$[\phi]^T [M] [\phi] = \begin{bmatrix} \diagdown & & \\ & M_j^* & \\ & & \diagdown \end{bmatrix} = \text{Generalized mass matrix}$$

$$[\phi]^T\,[C]\,[\phi] = \left[\diagdown\; 2\beta_j\,\omega_j\,M_j^* \;\diagdown\right] \quad = \quad \text{Generalized damping matrix} \qquad \text{(Eq. 3.7.2-3)}$$

$$[\phi]^T\,[K]\,[\phi] = \left[\diagdown\; \omega_j^2 M_j^* \;\diagdown\right] \quad = \quad \text{Generalized stiffness matrix}$$

where:

β_j = Damping ratio for the jth mode

ω_j = Undamped circular natural frequency of the jth mode

The undamped, circular natural frequencies are calculated by solving the following homogeneous equation:

$$[K] - \left[\diagdown\; \omega_j^2 \;\diagdown\right][M] = 0 \qquad \text{(Eq. 3.7.2-4)}$$

and the mode shape matrix for the jth mode is obtained from:

$$\left[K - \omega_j^2\,M\right]\left\{\phi_j\right\} = 0 \qquad \text{(Eq. 3.7.2-5)}$$

Knowing the damping value of each member of the structure (refer to Table 3.7-1) and computing the strain energy associated with each mode shape, composite modal damping values are calculated following the procedure shown in Section 3.7.2.15.

Premultiply Equation 3.7.2-1 by $[\phi]^T$ and substitute Equation 3.7.2-2 into Equation 3.7.2-1. Equation 3.7.2-1 then becomes the following.

$$\left[\diagdown\; M_j^* \;\diagdown\right]\left\{\ddot{q}\right\} + \left[\diagdown\; 2\beta_j\omega_j M_j^* \;\diagdown\right]\left\{\dot{q}\right\} + \left[\diagdown\; \omega_j^2\, M_j^* \;\diagdown\right]\{q\} \qquad \text{(Eq. 3.7.2-6)}$$

$$= -\,[\phi]^T\;[M]\;[T]\;\left\{\ddot{U}_g\right\}$$

For the case of one component of the ground motion $\ddot{U}_g$,

$$[T]\left\{\ddot{U}_g\right\} = \ddot{U}_g\;\{I\} \qquad \text{(Eq. 3.7.2-7)}$$

where:

$\{I\}$ = Vector whose elements are 1.0 for the degrees of freedom corresponding to the input direction of the ground motion; 0 for remaining degrees.

let:

$$[\phi]^T \quad [M] \quad \{I\} = \left\{\gamma_j M_j^*\right\} \qquad \text{(Eq. 3.7.2-8)}$$

where γ_j is the participation factor for the jth mode. Equation 3.7.2-6 then represents a set of N uncoupled modal equations, each of which can be written as:

$$\ddot{q}_j + 2\beta_j\omega_j\dot{q}_j + \omega_j^2 q_j = -\gamma_j \ddot{U}_g \qquad \text{(Eq. 3.7.2-9)}$$

where:

j = 1, 2, 3, ……, N

N = number of modes included (up to 33 Hz)

Each equation is solved using a step-by-step numerical integration technique of the following convolution integral (also called Duhamel integral) for zero initial conditions:

$$q_j(t) = -\frac{\gamma_j}{\omega_j\sqrt{1-\beta_j^2}} \int_o^t \ddot{U}(\tau)\, e^{-\beta_j\omega_j(t-\tau)} \sin\left[\omega_j\sqrt{1-\beta_j^2}\,(t-\tau)\right] d\tau \qquad \text{(Eq. 3.7.2-10)}$$

Once the uncoupled equations of motion have been solved for the generalized coordinates, q_j, at any instant of time, the relative displacements of the system $\{x_i\}$ are computed by the following equation.

$$x_i(t) = \sum_{j=1}^{N} \phi_{ij} q_j(t) \qquad \text{(Eq. 3.7.2-11)}$$

Any other responses of interest are determined from the above displacement response.

3.7.2.1.2 Response Spectrum Method: In this method, the base excitation of a structure is specified in the form of response spectrum curves (either acceleration, velocity, or displacement). In the project, acceleration response spectra have been used. The maximum modal displacement response for the jth mode, $q_{j,\,max}$, is directly obtained thus:

$$q_{j,\,max} = \gamma_j S_{aj}/\omega_j^2 \qquad \text{(Eq. 3.7.2-12)}$$

where:

ω_j = Frequency of jth mode

S_{aj} = Spectral acceleration corresponding to frequency, ω_j

γ_j = Participation factor for the jth mode (refer to Section 3.7.2.1.1).

Spectral acceleration, S_{aj}, also corresponds to a modal damping value, D_n, as discussed in Section 3.7.2.15. The maximum structural displacement response at any node, i, due to jth mode is then:

$$x_i = \phi_{ij} \; q_{j,\ max} \qquad \text{(Eq. 3.7.2-13)}$$

in which ϕ_{ij} = mode shape vector for jth mode. Other quantities of interest such as accelerations and member forces are computed in a straightforward manner.

Since all the modal response maxima do not occur simultaneously, the total response is obtained by a modal combination technique that involves probablistic considerations as discussed in Section 3.7.2.7.

3.7.2.2 Natural Frequencies and Response Loads.

3.7.2.2.1 Natural Frequencies: The two-step Finite Element Method (FEM) for SSI is an appropriate basis for seismic design; however, the natural frequencies obtained through the fixed-base models used in the two-step solution do not reflect the soil/structure interaction modes. Therefore, the natural frequencies summarized in Tables 3.7-3 through 3.7-6 are derived from elastic half-space (EHS) SSI analyses instead of the design-basis two-step FEM in order to represent the frequencies of the dominant modes corresponding to soil/structure interaction.

3.7.2.2.2 Structural Responses: Acceleration, bending moment, and shear force responses for the RCB and the MEAB are given on Figures 3.7-19A through 3.7-31. Structural responses are the envelopes of the response obtained from each analysis (upper-bound, average and lower-bound soil properties). Some geometric coupling is evident, accordingly, excitation in one horizontal direction (e.g., east/west, may produce significant responses [acceleration, bending moment, etc.] in the other horizontal direction [north/south]). Therefore, all codirectional responses due to individual analysis in each direction (east/west, north/south, and vertical) are combined by the square root of the sum of the squares (SRSS).

Floor seismic acceleration response spectra at selected locations are also based on codirectional combinations by SRSS. Procedures for the development of these spectra are discussed in Section 3.7.2.5.1.

3.7.2.3 Procedure Used for Modeling. For the first-step SSI analysis, the structures are represented by 2D Plane Strain finite element models which are coupled with the soil model in the 2D LUSH analysis to obtain the translational and rotational interaction base motion at the foundation level. A detailed discussion of soil/structure modeling is presented in

Section 3.7.2.4. For the second-step seismic analysis of the superstructures, three-dimensional, lumped-parameter mathematical models are constructed to represent each of the following structures:

1. Reactor Containment Building

2. Mechanical-Electrical Auxiliaries Building

3. Fuel Handling Building

4. Diesel Generator Building

For other Category I structures which are not included within the site cross-sections used in the two-step FEM analyses, the seismic analysis is performed though separate lumped-parameter models where the SSI is represented by either the EHS method or the free-field input motion amplified to account for SSI.

For a summary of the methods of seismic analysis for the various structures refer to Table 3.7-2. Mathematical lumped-parameter models for five major Category I structures, the RCB, MEAB, FHB, DGB and ECWIS are shown in Figures 3.7-16, 3.7-17, 3.7-17a through 3.7-17c.

In a lumped parameter model, the structure is represented by beam elements linked to nodal points at selected elevations where masses are lumped to represent floor weights, walls, and major equipment. The beams connecting those lumped masses are assumed weightless and elastic, representing the stiffness of walls between the lumped-mass nodal points. The foundation mat supporting the beam elements is also represented by a lumped mass, and is considered fixed except in the torsional degree of freedom for which a spring equivalent to the torsional stiffness between the structure and soil is provided. This torsional stiffness is computed by the following expression (Ref. 3.7.2-3):

$$K_\theta = \frac{16}{3} G r_o^3 \qquad \text{(Eq. 3.7.2-14)}$$

where:

G = Shear modulus of soil

r_o = Radius of circular foundation mat, and equal to $\sqrt[4]{\frac{16cd\left(c^2 + d^2\right)}{6\pi}}$,

for rectangular foundation mat where c and d are width and length of foundation, respectively.

Two types of mathematical models have been considered: one model subject to horizontal excitation, and the other model subject to vertical excitation. In the model for vertical excitation, only the vertical degrees of freedom are retained and others are eliminated by static condensation technique.

The lumped mass at each nodal point consists of dead load of the floor slab plus equipment and other permanent loads, and half the walls and columns above and below the floor. All six (three translational and three rotational) degrees of freedom are defined at each nodal point. Accordingly, translational and rotational masses about all three orthogonal axes are considered.

It is not practical to represent all equipment, piping or components in the general or primary mathematical model for the entire structure. Some equipment, piping or components which contribute significantly to overall structural response are directly represented as coupled submodels in the primary system model. Others have been classified as subsystems and are not specifically represented in the primary model but are included in the lumped masses of the model. Modeling, analytical techniques, and other aspects of subsystem analysis have been discussed in Section 3.7.3.

Wherever possible, the resonance condition of structures, systems and components and their supports are eliminated. However, if this is not possible, structures, systems and components are analyzed and designed for the peak of the response spectra.

The dynamic decoupling of systems from subsystems is based on the following criteria:

(1) $R_m \leq 0.01$

(2) $0.01 < R_m \leq 0.1$ (Eq. 3.7.2-15)

$R_f \geq 1.25$ or $R_f \leq 0.8$

where:

R_m = Effective mass ratio of subsystem to system

R_f = Natural frequency ratio of subsystem to system

The axial area, effective shear area, and the area moment of inertia of the beam elements linking the nodal points are calculated from the configurations of the walls and columns between the floors. Only the concrete walls and columns which extend from one nodal point to another are considered to contribute to the cross-sectional properties of the beam elements.

Hydrodynamic effects due to contained liquids are represented by lumped masses added to the model (for impulsive forces) and by additional oscillators, consisting of mass points and spring constants (for convective forces). The hydrodynamic effects are described in Reference 3.7.2-4.

3.7.2.4 Soil/Structure Interaction. Detailed SSI analyses are performed for Category I structures to account for the effect of the structures on the free-field motion and develop the input motion to be applied at the foundation base level of each structure. The FEM of analysis is used to develop mathematical models of the structures together with the adjacent and underlying soil media and perform the first- step of the SSI analysis. Detailed description of the first-step analyses is presented in Reference 3.7.2-12. From these analyses the accelerograms of the translational and rotational response at the soil/foundation interface are obtained. These accelerograms are used as the foundation base input motion for the second-step SSI analyses which are performed with the more detailed, fixed-base lumped-parameter models of the structures. The accelerograms contain the modifying effects due to the SSI and the interaction between the structures defined in the combined soil/structures models illustrated in Figures 3.7-18, 3.7-18A, 3.7-18B and 3.7-18C. The procedures used for these analyses are described in Sections 3.7.2.4.1 through 3.7.2.4.3.

Special Conditions: The two-step soil/structure interaction analyses described above, followed by the calculation of floor response spectra from the time-history acceleration response within structures is the seismic design basis for the STPEGS. This is referred to as the two-step FEM for SSI analysis. The corresponding response spectra calculated by the two-step FEM for SSI have been compared to the spectra calculated by the elastic half-space (EHS) method for SSI (Ref. 3.7.2-11). The comparison indicates that the two-step FEM spectra are conservative for the frequency range relevant to the seismic design and/or qualification of structural elements, equipment and components (i.e., above 4 Hz). The zero period acceleration values and the peak spectral responses obtained from the EHS spectra are lower or equal to the corresponding values from the FEM spectra. The only significant differences are detected in the low frequency range, mainly below 4 Hz, where the EHS spectral response for horizontal directions in some buildings is higher than the corresponding FEM spectral response. This difference is significant only for the RCB, is relatively insignificant for the FHB and the DGB, and it is essentially nonexistent for the MEAB (Figures 3.7-50 through 3.7-53). Therefore, the response spectra calculated by the two-step FEM for SSI are considered to be an adequate seismic design basis for the STPEGS subject to verification that the limited effects of the EHS-augmented spectra do not affect the seismic designs and/or qualifications established from the design-basis spectra. The specific difference detected in spectral response within the low frequency range is suitable for systematic assessment by natural-frequency segregation of the items susceptible to the higher seismic response developed in the identified frequency range of concern. The limited number of low-frequency items identified as affected by the EHS-augmented spectra are individually evaluated in accordance with the project design criteria to either demonstrate sufficient margin in their existing seismic design or qualification, or establish the need for reanalysis or requalification based on the EHS-augmented spectra.

The two-step FEM for SSI analysis, when implemented in the time domain utilizing decoupled fixed-base structural models as in the case of the STPEGS, is recognized to be (l) susceptible to potential under-representation of the soil/structure interaction effects on the spectral response, and (2) artificially sensitive to structural configuration with attendant over-representation of the spectral response within the structural frequency range. In order to address both of these potential limitations associated with the two-step FEM for SSI, consistent with the Standard Review Plan (SRP) Section 3.7.2. item II.4.a, a confirmatory-basis set of response spectra has been developed for the STPEGS. These spectra consist of the envelope of the response spectra obtained from the previously performed EHS solution and a single-step FEM solution. Both of these solutions are devoid of the structural decoupled models that introduce the potential limitations ascribed to the two-step FEM for SSI.

The EHS solution utilizes the soil impedances (springs and dampers) developed by the geotechnical consultant as described in Reference 3.7.2-13, and in- volves application of the free-field surface ground motion as direct input without any reduction to account for embedment of structures. The equivalent springs and dampers are a frequency-independent mechanical analog of the foundation impedances derived from elastic-half-space theory, and are based on average soil properties. The computed response spectra is subjected to a $\pm$ 15 percent frequency-based broadening to account for variations in soil and other properties in lieu of a parametric study involving upper- and lower-bound soil properties.

The single-step FEM solution utilizes for SSI analyses the computer program FLUSH described in Reference 3.7.2-8. The soil models are equivalent to the models developed by the general procedures described in Sections 3.7.2.4.1, 3.7.2.4.2, and 3.7.2.4.3. The same cross-sections defined in Figures 3.7-18, 3.7-18A, 3.7-18B, and 3.7-18C, plus two additional E-W cross-sections across the FHB and the DGB are used. The input accelerograms are equivalent to those described in this Section to define input motion at the base of the idealized soil model. The fundamental difference with respect to the previous two-step FEM is that the single-step FEM solution relies on a single transient analysis performed with 3D lumped-parameter mathematical models that permit detailed representations of the structures coupled to the 2D finite element model of the underlying soil, and the analytical solution is thus free of the structural decoupled models involved in second-step analyses.

The resultant confirmatory-basis spectra will not be used as the bases for seismic analyses and designs, and will not be released as a project design basis document. The purpose and use of the confirmatory-basis spectra are limited to the following:

1. To fulfill the provisions of the SRP Section 3.7.2, item II.4.a, pertaining to comparison of seismic response generated by EHS and by finite boundaries (FEM) methods for SSI. The approach elected corresponds to the envelope of the results of the two methods indicated as an acceptable operation in the SRP. The resultant enveloped spectra, herein referred to as confirmatory-basis spectra, are used to establish that the original STPEGS design-basis floor response spectra is (1) in general conservative, except for the limited under-representation with respect to the EHS spectra in the low frequency range, and (2) need not be revised as a function of structural configuration since the sensitivity to structural configuration is artificially introduced and is nullified by the over-conservatism of the original two-step FEM spectra.

2. Confirmation that the original design-basis spectra for the MEAB computed by two-step FEM based on original building configurations, as well as an existing version of the design spectra revised to reflect configurational changes in the MEAB based on two-step FEM, are conservative in the structural frequency range and are not affected by the EHS-augmented spectra. Therefore, since both the original and the configuration-revised spectra for the MEAB are higher than the confirmatory-basis spectra, analyses and designs based specifically and entirely on either of the two spectra are adequate. A detailed listing of seismic analysis methods is presented in Table 3.7-2.

3. Justification of specific instances where limited departures from the STPEGS design-basis floor response spectra are identified to exist in a completed seismic analysis, design

and/or qualification. These departures will be selectively considered only in cases where the over-conservatism of the design-basis spectra has been demonstrated and is valid to demonstrate the adequacy of the completed design with respect to the confirmatory-basis spectra. Table 3.7-8 addresses the specific cases where confirmatory-basis spectra are used.

The resultant confirmatory basis spectra are determined to be in reasonable agreement with those of the two-step FEM analyses, and thus fulfill the objective of demonstrating the general conservatism of the STPEGS design-basis spectra.

3.7.2.4.1 Development of Base-Level Motions: The base-level motions are those motions which, when input to the base of the soil model in the free-field, will produce the design response spectra at the finished grade level. Wave propagation procedures for the simulation of a free-field condition (shear beam analogy) have been used to obtain base-level motions as follows:

1. The idealized soil profile is modeled for the average soil properties. The artificial accelerogram whose spectra envelops the design response spectra (Section 3.7.1.2) is input at finished grade level in a deconvolution analysis to obtain the compatible base-level motion. The deconvolution analysis is made using the computer program TRIP (Ref. 3.7.2-10), which employs finite element techniques similar to those used in computer program LUSH (Ref. 3.7.2-5). Strain-dependent soil properties used in the TRIP analysis are obtained using the computer program SHAKE (Ref. 3.7.2-6). In the deconvolution analyses, it is required that the peak accelerations in the free-field at the foundation levels of Category I structures be not less than the values prescribed for the OBE and SSE (Section 3.7.1.1). This requirement is achieved by increasing, whenever necessary, the amplitude of the design input motion at the finished grade level.

2. The process described in paragraph 1, above, is repeated using both upper bound and lower-bound soil properties. The bounds of the properties are selected so that they cover the range of the properties measured in field and laboratory tests. In addition, the bounds are extended beyond the measured range, whenever necessary, to comply with requirements discussed in the following paragraph.

3. As part of the deconvolution analyses, the motions and response spectra in the free-field at finished grade and the foundation levels of Category I structures are computed and examined. It is required that the envelope of the three foundation-level spectra, obtained from deconvolution using average, upper-bound, and lower-bound properties, provides response spectral values that at finished grade are not less than the design response spectra and at the foundation level are in general not less than 60 percent of the design response spectra specified for the free-field at the finished grade level.

The base motions obtained by deconvolution for average, upper-bound, and lower-bound soil properties are used individually as input to finite element soil/structure models having the same (consistent) sets of soil properties.

The base motions are also used as input to finite element analysis of the idealized soil profile using computer program LUSH (Ref. 3.7.2-5). This analysis provides the means to establish finite element thicknesses and interpolation control numbers for use in program LUSH. The analysis provides verification that the motions at the finished grade and foundation levels in the free-field are reproduced using the finite element procedure. The response spectra of the motions at finished grade and foundation levels resulting from the analysis of the idealized soil profile are compared with the NRC criteria in Section 3.7.1.2.

Table 3.7-9 identifies the maximum relative displacement (other than at the base-level) due to earthquake and settlement among the principal power block structures. As stated in Sections 3.7.3B.8 and 3.7.3B.9, the effect of maximum relative displacements is included in the analysis of systems which interconnect structures. For piping systems, the analysis is in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III. The procedure used is in compliance with SRP Section 3.7.3.

3.7.2.4.2 Soil and Structural Modeling: The soil strata adjacent to and beneath the structure are represented by finite 2D plane strain elements. Several parametric studies and site sensitivity analyses are performed prior to the development of the final finite element soil models. These consist of dynamic time-history analyses on finite element models representing the structures and the typical site geology. The results of the analyses are used to assist in determining the extent of soil to be included in the soil models, the depth of significant interaction effects, the frequency transmission characteristics of the soil, and the detail required in the modeling of the soil to accurately represent the interaction response characteristics.

The width and depth of the finite element soil model are selected so that the presence of the lateral boundaries and the rigid base boundary do not significantly affect the structural response. The lateral boundaries are sufficiently distant from the structures so that waves are absorbed by internal damping before they can reflect back to the structures. This is ensured by attaining or nearly attaining a free-field condition at the lateral boundaries. Free-field conditions are checked by comparing response spectra of motions adjacent to the lateral boundaries with response spectra obtained from analysis of the idealized soil profile. The depth of the model is selected by analyzing models of progressively increasing depths and selecting a depth so that: (1) the structures do not significantly affect soil response at or below the selected depth (i.e., soil response is essentially uniform along a horizontal plane at or below the selected depth); and (2) greater model depths do not cause significant changes in response at the structural foundations.

The simplified finite element structural models (which include sufficient details to represent the significant structural characteristics) are combined with the soil finite element models to provide the soil/structure systems required for the first-step SSI analysis.

All analyses are made considering a two-dimensional continuum in plane strain. In modeling structural systems with a two-dimensional plane strain model, the following assumptions are made:

1. The out-of-plane structural responses are negligible in comparison with the structural responses in the direction of excitation input.

2. The torsional motions of the ground are too small to be considered.

For adequate representation of site conditions and structural configurations, analyses are performed for three critical site cross sections. locations of these cross-sections are shown on Figure 3.7-18. The cross-sections are selected to provide representation of the significant buildings and site conditions in the two principal plant directions. This permits evaluation of the seismic response of Category I structures during horizontal motions in two orthogonal directions. Cross sections 1 and 2 shown on Figure 3.7-18 are also analyzed for a vertical base motion.

3.7.2.4.3 Analytical Procedures: The FEM of analysis is used to evaluate the response at the foundation base of the soil/structure system for the duration of the base motion. The dynamic properties of the soil strata are selected based on their variation with strain induced due to the application of the base motion. Strain-compatible modulus and damping values are used in each element representing the soil strata. The damping parameters for the systems and components of the structure are selected based on the data in Sections 3.7.2.15 and 3.7.1.3.

The computer program LUSH (Ref. 3.7.2-5) is used for the first-step SSI analyses. This method of analysis incorporates the direct solution of the wave propagation equations in the system. It permits the transmission of frequencies considerably higher than would normally be obtained with other finite element methods. Because of the plane-strain representation, analyses using LUSH are generally regarded as providing an extremely conservative assessment of structure-to-structure interaction effects on response (Refs. 3.7.2-8 and 3.7.2-9).

The finite element soil/structure models described in Section 3.7.2.4.2 are subjected to base-level motions obtained as described in Section 3.7.2.4.1. For cross sections 1 and 2 (refer to Figure 3.7-18), a set of three analyses is made using average, upper-bound, and lower-bound soil properties. The response spectra obtained from these analyses at the foundations of Category I structures are examined to see the effect of the soil property variations. These analyses show that variations in foundation response spectra due to soil property variations are reasonably predicted from the results of a finite element analysis using average soil properties, plus the results of free-field deconvolution studies for average, upper-bound, and lower-bound properties. Finite element analyses for cross section 3 are performed for average properties only, and a reasonably conservative envelope is constructed representing the foundation response for the full range of properties (Ref. 3.7.2-7).

3.7.2.5 Development of Floor Response Spectra. Floor acceleration response spectra are developed from the time-history response records at selected points within the structures. The time-history responses are first obtained from dynamic analyses with detailed mathematical models of structures subjected to the SSI foundation base motion as described in Section 3.7.2.4. Next, the time-history acceleration response within the structures is used as input for the analysis of simple oscillators equivalent to a single-degree-of freedom system with various natural frequencies over the range of interest (0.5 Hz to 33 Hz) for several specific damping ratios. The maximum acceleration response obtained for the simple oscillators is then plotted for each damping ratio as a function of the corresponding natural frequency of the oscillator to obtain the spectral response over the whole frequency range.

The frequency intervals within the 0.5 to 33 Hz range used in the spectral response calculation are not consistent with the intervals given in RG 1.122. Parallel calculations to verify the adequacy of the original response spectra have been performed (Ref. 3.7.2-11). The results indicate that the only significant discrepancy associated with the frequency interval is related to the sparseness of the intervals used for the calculation of spectral response at frequencies below 2.5 Hz. For the higher and most relevant frequency range the frequency intervals used are adequate and the original response spectra is conservative; refer to Figure 3.7-54 as a typical example. Accordingly, the anomaly in spectral response introduced by the sparse frequency intervals is confined to the relatively unimportant low-frequency range, and the resultant spectra are enveloped by the EHS spectra used for the comparison addressed in Section 3.7.2.4. Therefore, the low-frequency interval implications on the spectral response are analogous and bounded by the EHS implications, and are similarly dispositioned (i.e., the original seismic response spectra are an adequate seismic design basis for the STPEGS subject to verification of the seismic design and/or qualification of the limited number of items affected by the discrepant spectral response confined to the low frequency range).

3.7.2.5.1 Procedure for Development of Floor Response Spectra: From each time-history analysis (east/west, north/south, and vertical), response spectra at selected nodal points are generated for the respective translational (plane) direction (i.e., east/west, north/south, and vertical, only). Due to asymmetry in structures, response components are developed in orthogonal directions other than the direction of input. For the horizontal excitation, the resultant response component in the other orthogonal direction is insignificant. For vertical excitation, the east/west and the north/south horizontal components are also insignificant. Nevertheless, for the structures in which SSI analyses have been performed, the horizontal responses due to vertical excitation and vice-versa have been taken into account by SRSS.

Each response spectrum is also widened on the frequency axis in order to take into account any parametric variations in properties, such as shear modulus, damping, material, etc. This is discussed in Section 3.7.2.9.

In general, spectra for OBE have been computed for 1-percent, 2-percent, and 4-percent damping values, and for SSE for 2 percent, 4 percent, and 7 percent. Typical floor response spectra are presented in Figures 3.7-32 through 3.7-49(O). It is noted that Figures 3.7-49A through 3.7-49F represent the structural-configuration revised spectra for the MEAB as described in Section 3.7.2.4.

3.7.2.6 Three Components of Earthquake Motion. For each mathematical model, three separate analyses have been performed based on the directions of input earthquake motion, namely, east/west, north/south, and vertical. Total structural responses (forces, displacements and accelerations) have been obtained by taking the SRSS of the co-directional maximum responses at a particular point of the structure obtained from each analysis.

For example:

$$R_i = \sqrt{(R_{ix})^2 + (R_{iy})^2 + (R_{iz})^2} \qquad \text{(Eq. 3.7.2-16)}$$

where:

R_i = Total shear in the ith direction

R_{ix} = Shear in the ith direction due to x-earthquake only, etc.

3.7.2.7 Combination of Modal Response. For modal time-history analyses performed independently along each principal direction, the modal responses are combined algebraically for each time step. For modal response spectrum analyses, the modal responses are combined using the SRSS method implemented in accordance with RG 1.92.

3.7.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures. Non-Category I structures in proximity of Category I structures are checked to verify that during the extreme loading conditions of an SSE they do not collapse onto Category I structures. Interaction of seismic Category I piping with nonseismic Category I piping is described in Section 3.7.3A.13.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra. Compliance with NRC criteria regarding response spectral values at the foundation levels in the free-field required a wide variation in soil dynamic shear moduli in SSI analyses (Section 3.7.2.4.1).

In constructing response spectra, shifting of the peaks with respect to natural frequencies by a minimum percentage is introduced to account for the uncertainties associated with computed natural frequencies of the structure. In cases where analyses have been performed for upper-bound, average and lower-bound shear moduli of soil (Section 3.7.2.4.3), the frequency variation, $\underline{+}f_j$, is determined by taking the SRSS of a minimum variation of $0.05f_j$ and the individual frequency variation $(\Delta f_j)_n$, that is:

$$\Delta f_j = \sqrt{(0.05f_j)^2 + \sum(\Delta f_j)^2 n} \qquad \text{(Eq. 3.7.2-17)}$$

A value of $0.10f_j$ is used if the actual computed value of f_j is less than $0.10f_j$.

In cases where only one soil case is considered, the spectrum is shifted by at least ±15 percent of each frequency.

3.7.2.10 Use of Constant Vertical Load Factors. Constant vertical load factors are not used to obtain vertical response loads for the seismic design of Category I structures, systems, and components. Multimass dynamic analyses for both horizontal and vertical directions of excitation are performed as described in Section 3.7.2.1, and a combination of three component earthquake responses is made.

For subsystems within structures, when the floor response spectra are used to define vertical input motion and/or loads for the seismic qualification and/or design of equipment and components, the rigidity of the structural subsystem is taken under consideration. Parametric analyses have been performed to determine the minimum subsystem frequencies required to assure effectively-rigid subsystem behavior that justifies use of the floor vertical response spectra directly without any additional amplification to account for subsystem flexibility. The results from the parametric analyses indicate that for structural subsystems whose vertical natural frequencies are above 8 cps in the MEAB, 10 cps in the RCB, 12 cps in the FHB and 16 cps in the DGB, the effect of subsystem flexibility on the floor vertical response spectra is insignificant. These frequency criteria are implemented in the project design criteria as a basic design requirement satisfied either by the initial design or by subsequent stiffening of the structural subsystems.

3.7.2.11 Method Used to Account for Torsional Effects. The actual three-dimensional soil/structure system is idealized and approximated by two-dimensional plane-strain models in the first-step FEM analysis. Thus, the insignificant effects of torsional motion on the development of the super-structural foundation motion is neglected in the SSI analyses. However, in the second step FEM analysis for calculation of structural responses, the torsional effect has been incorporated in the three-dimensional lumped- parameter models by providing a torsional spring at the foundation base as discussed in Section 3.7.2.3.

Subsequent structural analyses for Category I structures have been performed which account for the effect of accidental torsion (5 percent eccentricity).

3.7.2.12 Comparison of Responses. Only one method of seismic analysis (Table 3.7-2) has been used for each structure; therefore, comparison of responses calculated by an alternative method has not been made. Most of the major Seismic Category I structures have been analyzed using the modal time-history method. The time-history method involves direct integration at each time step. Therefore, the time-phase relationships between various modal responses are taken into account, resulting in calculated structural responses that are more reliable and accurate than those obtained from the combination of modal maxima from the response-spectrum method. Thus, no comparison of responses was considered to be necessary.

3.7.2.13 Methods for Seismic Analysis of Dams. There are no Category I dams associated with STPEGS. A discussion of the nonseismic Category I cooling lake dam and diversion dike is provided in Section 2.5.5.2. Seismic analysis of the Essential Cooling Pond (seismic Category I) is discussed in Section 2.5.5.

3.7.2.14 Determination of Seismic Category I Structures Overturning Moments. Overturning moments due to seismic effect on Category I structures are determined directly from the time-history analysis as described in Section 3.7.2.1.

3.7.2.15 Analysis Procedure for Damping. Equivalent modal damping is evaluated using the stiffness weighting technique as per Reference 3.7.2-2. This is an approximate method for determining modal damping by weighing the damping associated with the individual components, according to the strain energy stored in each component. Concrete structures, steel structures and systems, and foundation materials have inherently different damping properties, and the effective damping in any vibration mode of the total system depends upon the degree of participation of these components in the modal response.

The formula used to find the equivalent modal damping ratios for the natural modes of a structure having composite materials or substructures with different damping ratios is as follows:

$$D_n = \frac{\sum_{i=1}^{m} d_i S_{ni}}{S_n} \qquad \text{(Eq. 3.7.2-18)}$$

where:

D_n = Critical damping ratio for the nth mode

d_i = Material damping ratio for the ith structural component

S_{ni} = Strain energy of the ith structural component in the nth mode

S_n = Total strain energy of structure in the nth mode

m = Number of structural components

Other methods of computing equivalent modal damping for a soil-structure system are the "generalized equivalent modal damping technique" per Reference 3.7.2-14 and the "dissipating energy technique" per Reference 3.7.2-15.

3.7.3A Seismic Subsystem Analysis, Balance of Plant (BOP, exclusive of NSSS)

3.7.3A.1 Seismic Analysis Methods. For piping systems in the balance-of-plant (BOP) scope, the dynamic analyses are performed using the response spectrum method. Analyses by equivalent static load method is also used for design of piping systems which can be represented by a simple model to produce conservative responses.

The methods used for design of seismic Category I piping systems are as follows:

1. For ASME B&PV Code, Section III Class 1 piping, the response spectrum method is used for all piping sizes.

2. For ASME Section III Class 2 and 3 piping, the response spectrum method is used for piping of 2-1/2-in. nominal size and larger and for high-energy piping 2-in. nominal size and smaller that requires pipe break postulation. Remaining 2-in. nominal size and smaller piping is analyzed by either the simplified method or the static seismic method or the response spectrum method.

3. For ANSI B31.1 piping in seismic Category I buildings, the response spectrum method is used for high-energy piping that requires pipe break postulation. For piping that does not require pipe break postulation, either the simplified method or the static seismic method or the response spectrum method is used.

3.7.3A.1.1 Response Spectrum Method: The piping system is divided into series of finite elements to perform the dynamic analysis. The stiffness of each element is computed directly by generating the expression for the total strain energy in the element, including energy due to torsion.

The element stiffness matrix represents the set of loads necessary to produce a unit deflection while keeping all other degrees of freedom fixed at zero. Once these loads are computed for each element, the stiffness matrix for the complete piping system is obtained.

The equation of motion for the multidegree-of-freedom subsystem will be essentially the same as Equation 3.7.2-1, discussed in Section 3.7.2.1.1. For a single-degree-of-freedom system with displacement relative to base, x, mass, M, damping, C, and stiffness, K, the corresponding equation of motion is:

$$M\ddot{x} + C\dot{x} + Kx = -M\ddot{y} \qquad \text{(Eq. 3.7.3A-1)}$$

where:

$\ddot{y}$ = Absolute acceleration of base

Modal analysis technique is also discussed in Section 3.7.2.1.1, and Equation 3.7.2-9 is derived. The only difference here is the technique of calculating critical damping ratio, β_j, of jth mode. For subsystem analyses, a fraction of critical damping is assigned to each mode. It is not necessary to identify or evaluate individual modal damping coefficients.

Equation 3.7.3A-1 divided by M gives

$$\ddot{x} + \frac{c\dot{x}}{M} + \frac{Kx}{M} = -\ddot{y} \qquad \text{(Eq. 3.7.3A-2)}$$

After substitution, the equation reduces to

$$\ddot{x} + 2\beta\omega x + \omega^2 x = -\ddot{y} \qquad \text{(Eq. 3.7.3A-3)}$$

The following terms are defined:

$\beta = \dfrac{c}{2\omega M}$ = critical damping ratio for the single-degree-of-freedom system

$\omega = \sqrt{K/M}$ = natural frequency of the system

Equation 3.7.3A-3 is uncoupled for each mode, and can be solved as a single-degree-of-freedom system and all modes are independent of each other.

Contributions from all significant modes are combined in accordance with the RG 1.92.

3.7.3A.1.2 Equivalent Static Load Method: For piping systems, the analysis by equivalent static load method is performed utilizing the finite element computer programs. In the analysis, a constant acceleration in each global direction is applied to the piping system. The constant acceleration is obtained by multiplying the peak acceleration of the applicable floor response spectra by a factor of 1.5. Pipes are supported such that the piping stresses are kept within the code allowable limits. The seismic loads obtained by this method are included in the support designs.

The equivalent static load method is also used for design of cable tray supports and heating, ventilating, and air conditioning (HVAC) duct supports. A multiplication factor of 1.0 is applied to the peak spectral accelerations of applicable floor response spectra which are used to define the equivalent seismic load in each principal direction. Dynamic analyses using the modal response spectra method were performed for typical support systems to justify the use of the factor 1.0.

3.7.3A.1.3 Simplified Method: The simplified method involves the use of appropriate and comprehensive charts and tabulations to determine the piping spans, support loads, and types of supports. The seismic loads used in the design are obtained by using the concept of equivalent static load method. Piping spans are chosen to ensure that the piping stresses are within the code allowable limits.

3.7.3A.2 Determination of Number of Earthquake Cycles. The total number of significant earthquake cycles for the design of seismic Category I structures, systems and components is determined as a product of the number of postulated seismic events and the number of significant earthquake cycles per event.

As stated in Section 2.5.2.6, the duration of strong motion associated with the postulated SSE would be less than 5 seconds for which the number of significant cycles would be approximately two or three. To provide a conservative design basis, a minimum of ten maximum stress cycles per seismic event (one SSE and five OBEs) is selected.

3.7.3A.3 Procedure Used for Modeling.

3.7.3A.3.1 Mathematical Model for Piping Systems: Modeling procedures for subsystems have been discussed in Section 3.7.3A.1.1.

The preparation of a mathematical model for piping dynamic analysis is based on the following guidelines:

1. The piping system is modeled as a series of finite elements with masses lumped at certain nodal points.

2. The mass points are selected judiciously so that their locations coincide with the locations of large valves and supporting hangers.

3. The straight piping between the mass points is divided into a large enough number of elements to obtain a good approximation of all piping frequencies and mode shapes below 33 Hz.

3.7.3A.3.2 Modeling Procedure for Cable Tray Supports: A 3D static finite element analysis or a manual calculation, depending on the complexity of the system, is used to design the cable tray hanger. The finite element models, which represent the cable tray hangers, and transverse and longitudinal bracings, are simulated by beam elements interconnected with rotational springs at points. The dead loads, live load and seismic loads are applied simultaneously at the centers of horizontal members.

3.7.3A.3.3 Modeling Procedure for HVAC Ducts and Supports: Design of HVAC ducts and hangers is based on the equivalent static method. The duct response is determined by considering the beam deformation mode (resulting from the vertical, transverse and longitudinal restraints) and the sheet deformation mode (resulting from the stiffener effect).

3.7.3A.4 Basis for Selection of Frequencies. In the dynamic analysis, fundamental frequencies of subsystems and equipment are calculated based on the mass and stiffness characteristics of the systems. The seismic accelerations which the system must withstand are then determined from the applicable floor response spectra.

Three ranges of dynamic behavior of systems that have been considered for the magnitude of the seismic acceleration are:

1. In cases where the system is rigid relative to the structure, the maximum acceleration of the system approaches the low-period region of the floor response spectra.

2. In cases where the equipment is very flexible relative to the structure, the internal distortion of the structure is unimportant and the system behaves as though supported on the ground.

3. In cases where the periods of the system and supporting structure are nearly equal, resonance occurs and is taken into account.

Rigid systems are normally considered, by definition, when the natural frequencies are greater than the nominal value of 33 Hz. However, lower frequency limits may be used to establish rigid behavior in specific cases as determined from the dynamic response characteristics of the systems and the applicable floor response spectra.

3.7.3A.5 Use of Equivalent Static Load Method of Analysis. The use of equivalent static load method is discussed in Section 3.7.3A.1.2.

3.7.3A.6 Three Components of Earthquake Motion. The subsystem and equipment responses have been determined using the modal response spectrum analyses. The combination of modal responses from unidirectional analyses are performed by methodology that is in accordance with RG 1.92. The total response due to three directional excitation is then obtained by using the SRSS method or the component factor method (1,0.4 and 0.4) for the combination of co-directional responses from each excitation. The component factor method is not used for piping analysis.

3.7.3A.7 Combination of Modal Responses. For seismic Category I components in the BOP scope, the combination of modal responses for the response spectrum analyses is performed by the SRSS implemented in accordance with RG 1.92.

3.7.3A.8 Analytical Procedures for Piping. Analytical procedures for piping are discussed in Sections 3.7.3A.3.1 and 3.7.3A.9.

3.7.3A.9 Multiply-Supported Equipment and Components with Distinct Inputs. A dynamic response spectrum analysis is made assuming no relative displacement between support points. When a system is supported at different elevations in the same building with support points having different response spectra, or supported between buildings, a response spectrum which envelopes all the applicable response spectra has been used in the response spectrum analysis.

In certain cases, such as with auxiliary piping connected to the reactor coolant loop, multiple spectra have been used to reduce the excessive conservatism in applying enveloped spectra over the entire length of piping.

The effect due to differential seismic movements of piping supports in a piping system is included in the piping stress analysis in accordance with the requirements of NB-3650 in Section III ASME Code for Class 1 piping and NC/ND-3650 for Class 2 and Class 3 piping. The piping stresses, deflections and support loads induced by the differential seismic movements are computed using the most critical combination.

The effect of differential seismic movement of components interconnected between floors or buildings is considered statically in the integrated system analysis and in the detailed component analysis. For components, the differential motion is evaluated as a free-end displacement. 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 results of the dynamic inertia analysis and the static differential motion analysis, are combined by the SRSS method with due consideration for the ASME classification of the stresses.

3.7.3A.10 Use of Constant Vertical Static Factors. Whenever it is justified, constant vertical load factors are used as vertical response loads for subsystems, instead of multimass dynamic analyses. This procedure is adopted for both rigid and flexible components. Zero period accelerations are used for rigid components. For flexible components or for components with unknown natural frequency, 1.5 times the load corresponding to the peak of the applicable response spectrum curve is used to qualify piping and supports in accordance with the piping stress analysis criteria.

3.7.3A.11 Torsional Effects of Eccentric Masses. The effect of eccentric masses, such as valves and valve operators, is considered in the seismic piping analyses. The eccentric masses are modeled in the system analysis, and the resultant torsional effects 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.

3.7.3A.12 Buried Seismic Category I Piping Systems and Tunnels. Traveling seismic waves cause distortion of the ground during an earthquake. A buried structure (piping or duct banks) is forced to conform, in general, to the strains and curvature developed in the soil medium in which it is placed. However, because of slippage between the structure and the soil medium and the local deformations between the two, the deformation of the structure will generally be less than that of the medium, and the assumption that there is no relative motion between the structure and the soil is appropriate for most practical cases.

The earthquake response in a soil medium is derived from the passage of various types of waves such as P (compression), S (shear), and R (Raleigh) waves. The resultant strains and curvatures in the soil medium are calculated based on the wave propagation velocities and the maximum ground particle velocities and accelerations due to the design earthquake.

Stresses in the buried structures are developed due to the imposition of the soil strain and curvature on the structural rigidity with due regard of the angles of incidence and the maximum response from each of the various types of waves. The responses obtained from each wave type are combined by the SRSS method. The general expressions for calculating the axial and bending stresses are as follows: (Ref. 3.7.3-10)

$$\sigma_a = E\sqrt{\left(\frac{V_p}{C_p}\right)^2 + \left(\frac{V_s}{2C_s}\right)^2 + \left(\frac{V_r}{C_r}\right)^2}$$

$$\sigma_b = ER\sqrt{\left(\frac{0.385A_p}{{C_p}^2}\right)^2 + \left(\frac{A_s}{{C_s}^2}\right)^2 + \left(\frac{0.385A_r}{{C_r}^2}\right)^2}$$

Where:

Where:	V	=	Soil particle velocity
	p,s,r	=	Subscripts refer to compression, shear and surface waves, respectively
	C	=	wave propagation velocity
	A	=	soil particle acceleration
	E	=	modulus of elasticity for the material of the structure
	R	=	distance from the neutral axis to the extreme fiber

Additional seismic stresses resulting from the effects of discontinuity and differential displacements at connections to buildings are obtained based on the formulations for beams on elastic foundations which are discussed in References 3.7.3-10 and 3.7.3-11.

3.7.3A.13 Interaction of Other Piping with Seismic Category I Piping. Where seismic Category I piping systems are in close proximity to non-seismic piping, the non-seismic pipes are restrained so that no failure of the seismic Category I system can occur.

Where seismic Category I piping is directly connected to nonseismic Category I piping, the seismic effects of the latter are prevented from being transferred to the seismic Category I piping by use of anchors or a combination of restraints; or when this is not practical, the interactive effects of the unrestrained portion of the non-seismic Category I piping are included in the analyses.

3.7.3A.14 Seismic Analysis for Reactor Internals. See Section 3.7.3B.

3.7.3A.15 Selection of Damping. For BOP piping analysis, the lowest damping value associated with the elements of the system is used for all modes. For components in seismic Category I buildings, the critical damping values used are in accordance with RG 1.61 and are included in Table 3.7-1.

The following Pressure Vessel Research Committee (PVRC) recommended damping values per ASME Code Case N-411 have also been used in the piping stress analysis.

Frequency Range Hz	Damping Value
0 to 10	5%
10 to 20	5% to 2% (linear reduction)
20 and Higher	2%

PVRC Damping values are used with the following conditions and limitations:

1. PVRC damping values are used for support optimization, for as-built reconciliation, and for new analysis to reduce piping stresses and support loads.

2. PVRC damping values are used only for response spectrum method analysis. PVRC damping values are not used for the time history method analysis.

3. Piping stress calculations have used either PVRC damping values or RG 1.l damping values. A combination of PVRC and RG 1.l damping values are not used within the same analysis.

4. When PVRC damping values are used, it is verified that the clearance between the piping and other plant structures, components and equipment is adequate so that the piping does not adversely interact with them due to increased motion, and the mounted equipment can withstand the increased motion.

For design of HVAC duct and supports, damping values of 2% (OBE) and 4% (SSE) are used. For cable tray supports the applicable damping values are selected as follows:

1. For frame systems with bolted moment connectors, 7% for the SSE.

2. For laterally and longitudinally braced systems constructed with cold-rolled thin metal struts, the damping values are obtained from Figure 3.7-55. The upper limits of damping given are based on generic dynamic testing of cable tray supports of the described type.

3.7.3B Subsystem Seismic Analysis, NSSS Scope

3.7.3B.1 Seismic Analysis Methods. Those components and systems that must remain functional in the event of the SSE are identified by applying the criteria of Section 3.2.1. The equipment is classified into three types according to its dynamic characteristics. The analysis methods used for the equipment also depend on these classifications.

The first type of equipment is classified as flexible. This equipment is characterized by several modes in the frequency range that could produce amplification of the base input motion. Because of these reasons, dynamic analyses were performed for these components using response spectrum analysis, integration of the uncoupled modal equations, direct integration of the coupled differential equation of motion, or nonlinear modal superposition.

The second type of equipment is classified as rigid. This equipment has a fundamental natural frequency that is sufficiently high (greater than 33 Hz) so that base input motions are not amplified. Such equipment is particularly suitable for static analysis as described Section 3.7.3B.1.7.

Finally, the third type of equipment is classified as limited flexible, with only one predominate mode in the frequency range subject to possible amplification of the input motion. The fundamental mode of this type of equipment is basically a translations bending mode at a frequency less than 33 Hz. The second mode is usually a rocking mode with a frequency greater than 33 Hz. Because of the simple response characteristics of the equipment, dynamic analysis techniques that account for multiple mode effects and closely spaced modes are not required. Therefore, this equipment was evaluated using static analysis methods as described in Section 3.7.3B.1.7.

3.7.3B.1.1 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 dashpots 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. Some typical modeling techniques are presented in Reference 3.7.3-3.

Equations of Motion

Consider the multidegree-of-freedom system shown in Figure 3.7-56. Making a force balance on each mass point r, the equations of motion can be written in the form:

$$m_r \ddot{y}_r + \sum^{i} C_{ri} \dot{u}_i + \sum^{i} k_{ri} u_i = 0 \qquad \text{(Eq. 3.7.3B-1)}$$

where:

m_r = the value of the mass or mass moment of rotational inertia at mass point r.

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.

$\dot{u}_i$ = 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 the displacement (rotation).

u_i = displacement (rotation) of mass point i relative to the base.

Since:

$$\ddot{y}_r = \ddot{u}_r + \ddot{y}_s \qquad \text{(Eq. 3.7.3B-2)}$$

where:

$\ddot{y}_s$ = absolute translational acceleration of the base.

$\ddot{u}_r$ = translational (angular) acceleration of mass point r relative to the base.

Equation (3.7.3B-1) can be written as:

$$m_r \ddot{u}_r + \sum^{i} c_{ri}\dot{u}_i + \sum^{i} k_{ri} u_i = -m_r \ddot{y}_s \qquad \text{(Eq. 3.7.3B-3)}$$

For a single degree-of-freedom system with displacement u, mass m, damping c, and stiffness k, the corresponding equation of motion is:

$$m\ddot{u} + c\dot{u} + ku = -m\ddot{y}_s \qquad \text{(Eq. 3.7.3B-4)}$$

3.7.3B.1.2 Modal Analysis:

Natural Frequencies and Mode Shapes

The first step in the modal analysis method is to establish the normal modes, which were determined by the eigensolution of Equation 3.7.3B-3. The right hand side and the damping term are set equal to zero. Thus, Equation 3.7.3B-3 becomes:

$$m_r \ddot{u}_r + \sum^{i} k_{ri} u_i = 0 \qquad \text{(Eq. 3.7.3B-5)}$$

The equation given for each mass point r in Equation 3.7.3B-5 can be written as a system of equations in matrix form as:

$$[M]\{\ddot{\Delta}\} + [K]\{\Delta\} = 0 \qquad \text{(Eq. 3.7.3B-6)}$$

where:

$[M]$ = diagonal 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, $d^2 \{\Delta\} / d^2 t$.

Harmonic motion is assumed and $\{\Delta\}$ is expressed as:

$$\{\Delta\} = \{\delta\} \text{ Sin } \omega t \qquad \text{(Eq. 3.7.3B-7)}$$

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 3.7.3B-6 and yield:

$$[K]\{\delta\} = \omega^2 [M]\{\delta\} \qquad \text{(Eq. 3.7.3B-8)}$$

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 3.7.3B-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 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:

$$k - \omega^2 m = 0$$

or: (Eq. 3.7.3B-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:

$$f = \frac{1}{2\pi}\sqrt{\frac{k}{m}} \qquad \text{(Eq. 3.7.3B-10)}$$

To find the mode shapes, the natural frequency corresponding to a particular mode, ω_n, can be substituted in Equation 3.7.4-8.

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 3.7.3-6 pp. 116-125):

$$\ddot{A}_n + 2\omega_n \; p_n \; \dot{A}_n + \omega_n^2 A_n = -\Gamma_n \ddot{y}_s \qquad \text{(Eq. 3.7.3B-11)}$$

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:

$$u_{rn} = A_n \phi_{rn} \qquad \text{(Eq. 3.7.3B-12)}$$

where:

ω_n = natural frequency of mode n in radians per second.

Γ_n = modal participation factor of mode n given by:

$$\Gamma_n = \frac{\sum^{n} m_r \phi'_{rn}}{\sum^{n} m_r \phi^2_{rn}} \qquad \text{(Eq. 3.7.3B-13)}$$

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 3.7.3B-11 is analogous to the equation of motion for a single degree-of-freedom system that will be developed from Equation 3.7.3B-4. Dividing Equation 3.7.3B-4 by m gives:

$$\ddot{u} + \frac{c}{m}\dot{u} + \frac{k}{m}u = -\ddot{y}_s \qquad \text{(Eq. 3.7.3B-14)}$$

The critical damping ratio of a single degree-of-freedom system, p, is defined by the equation:

$$p = \frac{c}{c_c} \qquad \text{(Eq. 3.7.3B-15)}$$

where the critical damping coefficient is given by the expression:

$$c_c = 2m\omega \qquad \text{(Eq. 3.7.3B-16)}$$

Substituting Equation 3.7.3B-16 into Equation 3.7.3B-15 and solving for c/m gives:

$$\frac{c}{m} = 2\omega p \qquad \text{(Eq. 3.7.3B-17)}$$

Substituting this expression and the expression for k/m given by Equation 3.7.3B-9 into Equation 3.7.3B-14 gives:

$$\ddot{u} + 2\ p\dot{u} + \omega^2 u = -\ddot{y}_s \qquad \text{(Eq. 3.7.3B-18)}$$

Note the similarity of Equations 3.7.3B-11 and 3.7.3B-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 critical damping ratio 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 is not appropriate for a slightly damped structure supported by a massive moderately damped structure. There are several methods which can be used to incorporate damping in a structural system.

One 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. A 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. A third method is to use the Rayleigh damping method based on computed modal energy distribution. In yet another method, the damping value for a given mode is derived from the calculation of the composite modal damping which is based on the distribution of energy in the structure for that mode.

3.7.3B.1.3 Response Spectrum Analysis: The response spectrum is a plot showing the variation in the maximum response (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 that base motion. The spectral acceleration, velocity, and displacement are related by the equation:

$$S_{a_n} = \omega_n \, S_{v_n} = \omega_n^2 \, S_{d_n} \qquad \text{(Eq. 3.7.3B-19)}$$

In addition to ground motion input spectra (Equation 3.7.3B-19), the response spectra at various support points (i.e., floor response spectra) are developed for use in design of subsystems located at various elevations.

3.7.3B.1.4 Integration of Uncoupled Modal Equations: This method can be separated into the following two basic parts:

1. Integration procedure for the uncoupled modal equation (Equation 3.7.3B-11) to obtain the modal displacements and accelerations as a function of time.

2. Using these modal displacements and accelerations to obtain the total displacements, accelerations, forces, and stresses.

Integration Procedure

Integration of these uncoupled modal equations is done by a 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, 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.

Total Displacements, Accelerations, Forces and Stresses

From the modal displacements and accelerations, the total displacements, accelerations, forces, and stresses can be determined as follows:

1. Displacement of mass point r in mode n as a function of time is given by Equation 3.7.3B-12 as:

$$u_{rn} = A_n \phi_{rn} \qquad \text{(Eq. 3.7.3B-20)}$$

with the corresponding acceleration of mass point r in mode n as:

$$\ddot{u}_{rn} = \ddot{A}_n \phi_{rn} \qquad \text{(Eq. 3.7.3B-21)}$$

2. The displacement and acceleration values obtained for the various modes are superimposed algebraically to give the total displacement and acceleration at each time interval.

3. 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.

3.7.3B.1.5 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 displacement and elemental stresses for the complete structure are calculated as functions of time.

The basic equations for the dynamic analysis are as follows:

$$[M]\{\ddot{x}\} + [c]\{\dot{x}\} + [K]\{x\} = \{F(t)\} \qquad \text{(Eq. 3.7.3B-22)}$$

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 preload 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 $-[M]\ \{\ddot{z}\}$ to the right hand side of the basic equation (Equation 3.7.3B-22); i.e.,

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = F - [M]\{\ddot{z}\} \qquad \text{(Eq. 3.7.3B-23)}$$

The vector $\{\ddot{z}\}$ is defined by its components z_i where i refers to each degree of freedom of system. z_i is equal to a_1, a_2, or a_3 if the ith degree of freedom is aligned with the direction of the base translational acceleration a_1, a_2, or a_3 respectively. $z_i = 0$ if the ith degree of freedom is not aligned with any direction of the base 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 3.7.3B-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 stage 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.

3.7.3B.1.6 Nonlinear Modal Superposition: In the nonlinear modal superposition method the nonlinearities are presented as pseudo force. The mass and stiffness matrices are calculated only once and the corresponding mode shapes and natural frequencies are associated with the linear system simulating the initial state of the undamped structure with no external force acting on it. This state of the structure is hereafter referred to as the reference state. During the time-history analysis, as the nonlinear behavior comes into action, the true frequencies and mode shapes change. The effect of the variation of the true frequencies and mode range from the original ones is represented by pseudo forces on the right hand side of the equation of the equation of motion.

The generalized equation of motion for a nonlinear structure is:

$$[M]\{\ddot{x}\} + [C_{nl}]\{\dot{x}\} + [K_{nl}]\{x\} = \{F\} \qquad \text{(Eq. 3.7.3B-24)}$$

where:

$[M]$	=	mass matrix
$[C_{nl}]$	=	nonlinear damping matrix, dependent upon velocity and displacement
$[K_{nl}]$	=	nonlinear stiffness matrix, dependent upon displacement
$\{\ddot{x}\}$, $\{\dot{x}\}$, $\{x\}$ and $\{F\}$	=	acceleration, velocity, displacement and applied force vector

let $$[C_{nl}] = [C] + [\overline{C}] \qquad \text{(Eq. 3.7.3B-25)}$$

and $$[K_{nl}] = [K] + [\overline{K}]$$

where [C] and [K] are the damping and stiffness matrices representing the reference state of the structure, $[\bar{C}]$ and $[\bar{K}]$ are the damping and stiffness matrices, dependent on velocity and displacement. Substitution of equation (3.7.3B-25) into equation (3.7.3B-24) gives:

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{F - F_{nl}\} \qquad \text{(Eq. 3.7.3B-26)}$$

where the pseudo-force vector is defined by:

$$\{F_{nl}\} = [\bar{C}]\{\dot{x}\} + [K]\{x\} \qquad \text{(Eq. 3.7.3B-27)}$$

The homogenous, undamped equation of motion representing the reference state of the structure is:

$$[M]\{\ddot{x}\} + [K]\{x\} = \{0\} \qquad \text{(Eq. 3.7.3B-28)}$$

Let $[\omega]$ and $[\phi]$ be the natural frequency and normalized mode shape matrix. The following transformation:

$$\{x\} = [\phi]\{q\} \qquad \text{(Eq. 3.7.3B-29)}$$

is substituted in equation 3.7.3B-26, resulting in the following uncoupled modal equations:

$$\{q\} + \left[2\varsigma_j\ \omega_j\right]\{\dot{q}\} + \left[\omega_j^2\right]\{q\} = \{Q\} - \{Q_{nl}\} \qquad \text{(Eq. 3.7.3B-30)}$$

where:

ζ_j		=	percentage of the critical damping for the jth mode
$\{Q\}$	$= [\phi]^T\{F\}$	=	generalized applied force vector
$\{Q_{nl}\}$	$= [\phi]^T\{F_{nl}\}$	=	generalized pseudo force vector

Arrays $\{q\}$, $\{\dot{q}\}$ and $\{\ddot{q}\}$ are the modal displacement, velocity and acceleration vector, respectively. The generalized pseudo-force vector is a function of displacement and velocity. For a given time step, it can be approximated by the Taylor series.

For a given time step, modal equations of motion are integrated analytically. Then the displacement and velocities of the nodes associated with the non-linear elements are calculated. This information is used to calculate the generalized pseudo-force vector and its time derivatives. Then the modal equations are integrated for the next time step.

3.7.3B.1.7 <u>Static Analysis - Rigid and Limited Flexible Equipment</u>: Rigid equipment and limited flexible equipment as defined in Section 3.7.3B.1 are generally analyzed using the static analysis method. This technique involves the multiplication of the total weight of the equipment or component member by a specified seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient was established on the basis of the excitation level that the component was expected to experience in the plant.

For rigid equipment, the seismic acceleration coefficients were compared with the high frequency (greater than 33 Hz) acceleration levels for the applicable response spectra developed for the plant to confirm the design analysis. The seismic acceleration coefficients for limited flexible equipment are compared with the acceleration levels from the applicable response spectra at the calculated fundamental natural frequency of the component. If the design seismic acceleration coefficients for either rigid or limited flexible equipment are exceeded by the actual plant acceleration levels, the design analysis is performed again at the actual level to confirm the equipment adequacy.

3.7.3B.2 Determination of Number of Earthquake Cycles. The OBE is conservatively assumed to occur five times over the life of the plant. A time history study has been conducted to arrive at a realistic number of maximum stress cycles per OBE occurrence for all Westinghouse systems and components.

This evaluation considered both the equipment and its supporting building structure as single-degree-of-freedom systems, which tend to produce a more uniform and unattenuated response than a complex, interacting system. The natural frequencies for the building and equipment are conservatively chosen to coincide.

As a result of this study, 10 maximum stress cycles for equipment for each OBE occurrence are used for fatigue evaluation of Westinghouse systems and components.

3.7.3B.3 Procedure Used for Modeling. Modeling technique is discussed in Section 3.7.3B.1.

3.7.3B.4 Basis for Selection of Frequencies. In the analysis of the Class 1 branch lines attached to the reactor coolant loop (including the surge line), the frequencies of these lines may be controlled if necessary to avoid the peak building frequencies and the lowest fundamental frequencies of the primary equipment, to maintain the equipment and support loads within allowable limits.

There is no specific design criteria which attempts to control the fundamental frequencies of NSSS equipment to be different from the forcing frequencies of the supporting structures. The effect of the equipment fundamental frequencies relative to the support structure forcing frequencies is, however, considered in the analysis of the NSSS equipment.

Three ranges of equipment/support behavior which affect the magnitude of the seismic acceleration are possible:

1. 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.

2. If the equipment is very flexible relative to the structure, the equipment will show very little response.

3. If the periods of the equipment and supporting structure are nearly equal, resonance occurs and must be taken into account.

Also, as noted in Section 3.7.3B.1, rigid equipment/support systems have natural frequencies greater than 33 Hz.

3.7.3B.5 Use of Equivalent Static Load Method of Analysis. This subject is discussed in Section 3.7.3B.1.7.

3.7.3B.6 Three Components of Earthquake Motion. The unidirectional responses obtained from unidirectional analyses as described in Section 3.7.3B.7 are combined using the SRSS methods to obtain the total response.

3.7.3B.7 Combination of Modal Responses. For seismic Category I components in NSSS scope, the method used to combine modal responses is described below. The total unidirectional seismic response for NSSS equipment is obtained by combining the individual modal responses using the SRSS 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. Combined total response for systems which have such closely spaced modal frequencies is obtained by adding to the SRSS of all modes the product of the responses of the modes in each group of closely spaced modes and a coupling factor, ε. This can be represented mathematically as:

$$R_T^2 = \sum_{i=1}^{N} R_i^2 + 2 \sum_{j=1}^{S} \sum_{k=M_j}^{N_j - 1} \sum_{\ell = k+1}^{N_j} R_k R_\ell \varepsilon_{k\ell} \qquad \text{(Eq. 3.7.3B-31)}$$

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

$\varepsilon_{k\ell}$ = Coupling factor with

$$\varepsilon_{k\ell} = \left\{1 + \left[\frac{\omega'_k - \omega'_\ell}{\beta'_k\omega_k + \beta'_\ell\omega_\ell}\right]^2\right\}^{-1} \quad \text{(Eq. 3.7.3B-32)}$$

and

$$\omega'_k = \omega_k\left[1 - (\beta'_k)^2\right]^{1/2} \quad \text{(Eq. 3.7.3B-33)}$$

$$\beta'_k = \beta_k + \frac{2}{\omega_k t_d} \quad \text{(Eq. 3.7.3B-34)}$$

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

There are two groups of closely spaced modes, namely 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 3.7.3B-31:

$$R_T^2 = \left[R_1^2 + R_2^2 + R_3^2 + \ldots\ldots + R_8^2\right] + 2R_2R_3\varepsilon_{23} + 2R_2R_4\varepsilon_{24} + 2R_3R_4\varepsilon_{34} + 2R_6R_7\varepsilon_{67}$$

(Eq. 3.7.3B-35)

3.7.3B.8 Analytical Procedures for Piping. Class l piping systems are analyzed to the rules of the ASME B&PV Code, Section III, NB-3650. When response spectrum methods are used to evaluate piping system supported at different elevations, the following procedures are used. The effect of differential seismic movement of piping supports is included in the piping analysis according to the rules of the ASME B&PV Code, Section III. According to ASME definitions, these displacements cause secondary stresses in the piping system.

In the response spectrum dynamic analysis for evaluation of piping systems supported at different elevations, spectra which envelope the floor response spectra corresponding to the applicable support locations are used. Westinghouse does not have in their scope of analysis any piping systems interconnected between buildings.

3.7.3B.9 Multiply Supported Equipment and Components with Distinct Inputs. When response spectrum methods are used to evaluate RCS primary components interconnected between floors, the procedures of the following paragraphs are used. The primary components of the RCS 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 envelope of the floor response spectra corresponding to the various support elevations.

Secondly, the effect of differential seismic movement of components interconnected between floors is considered statically in the detailed component analysis. Per ASME B&PV 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, per NB-3213.19.

The results of these two steps, dynamic inertia analysis and the static differential motion analysis, are combined absolutely with due consideration for the ASME classification of the stresses.

3.7.3B.10 Use of Constant Vertical Static Factors. This Section is not applicable (constant vertical static factors are not used in NSSS analysis).

3.7.3B.11 Torsional Effects of Eccentric Masses. Torsional effects of eccentric masses are discussed in Section 3.7.3A.11.

3.7.3B.12 Buried Seismic Category I Piping Systems and Tunnels. This section is not applicable.

3.7.3B.13 Interaction of Other Piping with Seismic Category I Piping. This Section is not applicable.

3.7.3B.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 the typical fuel assemblies is contained in Reference 3.7.3-7.

Fuel assembly lateral structural damping obtained experimentally is presented in Figure B-4 of Reference 3.7.3-7. The distribution of fuel assembly amplitudes decreases as one approaches the center of the core.

The CRDM are seismically analyzed to confirm that system stresses under the combined loading conditions, as described in Section 3.9.1, do not exceed allowable levels as defined by the ASME B&PV Code, Section III. 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 B&PV Code, Section III requirements.

3.7.3B.15 Analysis Procedure for Damping. The damping values given in Table 3.7-7 are used for the systems analysis of Westinghouse equipment. These are consistent with the damping values recommended in RG 1.61 except in the case of the primary coolant loop system components and large piping (excluding RPV internals) for which the damping values of 2 and 4 percent are used as established in testing programs reported in WCAP-7921-AR. CN-2985 The damping values for the CRDM and the fuel assemblies of the NSSS, when used in seismic system analysis, are in conformance with the values for welded and/or bolted steel structures (as appropriate) listed in RG 1.61. CN-2985

Tests on fuel assembly bundles justified conservative component damping values of 7 percent for OBE and 10 percent for SSE to be used in the fuel assembly component qualification. Documentation of the fuel assembly tests will be found in Gesinski and Chiang (Refs. 3.7.3-8 and 3.7.3-9).

The damping values used in component analysis of the 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 3.7-7 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 as described in WCAP 7427. CN-2985 These are encircled by a box section frame which is attached to 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 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 actual CRDM design uses seismic sleeves that engage holes in a seismic missile shield plate. The support system in the test rig is dynamically equivalent to the actual CRDMs. CN-2985

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 minimum measured damping is greater than 9 percent. The clearance in a typical upper seismic CRDM support is a minimum of 0.10 inches. CN-2985 The increasing damping with increasing clearances trend from the test results indicated that the damping would be greater than 8 percent for both the 1/2 SSE 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 the CRDM component analyses by response spectra techniques.

3.7.4 Seismic Instrumentation

3.7.4.1 Comparison with Regulatory Guide 1.12. The seismic instrument program takes exception to RG 1.12, Rev. 1. The program uses an alternate approach as follows:

1. The seismic instrumentation is a digital triaxial seismograph unit with programmable alarm, trigger, memory, recording and data retrieval capabilities and personal computer interface. The instrument is capable of providing time history acceleration data. The appropriate trigger condition will be selected to start data capture into solid-state memory or removable memory cards for later analysis. Settings for the instrument's pre-event memory and length of time that data is recorded will be selected so that the significant ground motion associated with the earthquake is recorded. The recorded information can be analyzed and displayed using a personal computer and software supplied with the machine. This software will display the measured response spectrum to be compared with the OBE and SSE response spectrum.

2. The Triaxial Seismic Trigger designed to monitor the acceleration at the Containment base slab has an actuation level adjustable over a minimum range of 0.01g to 0.03g, in lieu of the minimum sensitivity level of 0.005g specified in ANSI/ANS Standard 2.2, paragraph 5.4.1. Triggering levels below 0.01g are likely to produce spurious triggering due to normal plant vibrations.

3.7.4.2 Location and Description of Instrumentation. The seismic monitoring instrument is powered from the non-Class 1E 120 vac instrument bus. However, battery backup is capable of maintaining the instrument in a fully operational mode for at least one hour without plant power support.

The seismic monitoring provides all necessary functions, especially:

- Detection and permanent recording of seismic events.
- Prompt determination of the nuclear power plant seismic response necessary for the decision to shut down the plant.

The seismic instrument is located at the –37 foot level in the Unit 1 containment building tendon access gallery. This location has an existing calculated structural response spectrum.

In addition to the information in section 3.7.4.1.1, the seismic instrument has the following capabilities.

The instrument will be calibrated at predetermined intervals to ensure data accuracy. Backup battery replacement, periodic self-testing, inspections for damage, and checks for appropriate indications, as applicable, would be conducted at intervals to ensure continued satisfactory performance.

Seismic event information recorded in the seismograph will be available immediately. Data retrieval will be accomplished either by removing a computer disk or portable memory, or by transferring data directly via serial connection or other similar means to a personal computer. The recorded information will be retrieved and fed into a personal computer for processing. The event response ‘g’ spectrum will be compared with the OBE and SSE response spectrum allowing operations personnel to determine if the OBE has been exceeded.

3.7.4.3 Control Room Operator Notification. Control room indication of a seismic event will rely upon receiving an annunciator alarm in the control room from the trigger of the stand-alone instrument, so that acceleration data can be readily obtained from the stand-alone instrument. This data should be processed shortly after occurrence of an earthquake.

3.7.4.4 Comparison of Measured and Predicted Responses. The plant operators are provided with a procedure and criteria to review the accelerations recorded by the stand-alone instrument. The criteria consider system design and dynamic analyses in establishing the acceptable levels for continued operation.

Determination of exceedance of the Operating Basis Earthquake (OBE) consists of a check of the response spectrum and a check on the operability of the instrumentation. If the OBE is exceeded or significant plant damage occurs, both units will be shutdown unless plant walk-downs indicate plant damage precludes achievement of safe shutdown capability without corrective action. In the event safe shutdown is precluded, a plan for safe shutdown will be proposed by the South Texas Project to the Nuclear Regulatory Commission for approval.

Direct verification of the seismic responses of seismic Category I systems and components will not be performed. Information retrieved by the stand-alone instrument can be used to bound the impact of the siesmic recorded event on the other Seismic Category I structure, systems and components. The stand-alone instrument will allow comparison of actual response spectra to design response spectra at the location of the instrument. Measurements taken at one location cannot prove that accelerations at all other locations in the plant were less than design values. Nevertheless, the calculation techniques used to establish design response spectra were similar as to methods, assumptions and accuracy for all buildings. Therefore, establishing that the response at one building location did not exceed design values provides a strong basis for presuming that the siesmic responses at other locations in the plant were likewise bounded by design. Plant inspections and testing can be used to access the further capability of systems, structures and components for meeting safety functions.

REFERENCES

Section 3.7:

3.7.1-1 Not Used.

3.7.1-2 "Damping Values of Nuclear Plant Components", WCAP-7921 AR (May 1974).

3.7.1-3 Hall, W. T., and N. M. Newmark, "Seismic Design for Pipelines and Facilities", Journal of the Technical Council on Lifeline Earthquake Engineering of ASCE, No. TC1 (November, 1978), pp 91-107.

3.7.2-1 McDonnel-ECI ICES STRUDL DYNAL User's Manual, McDonnel Douglas Automation Company.

3.7.2-2 Whitman, R. V., Soil-Structure Interaction, Seismic Design for Nuclear Power Plants, ed. R. J. Hansen, The MIT Press (Cambridge, Massachusetts, and London, England), pp. 249.

3.7.2-3 Richart, F.E.,Jr., J. R. Hall, and R. D. Woods, Vibrations of Soils and Foundations, Prentice-Hall, Inc. (Englewood Cliffs, New Jersey, 1970), pp. 334-385.

3.7.2-4 Newmark, Nathan M., and Emilo Rosenblueth, Fundamentals of Earthquake Engineering, Prentice-Hall, Inc., 1970, Chapter 6.

3.7.2-5 Lysmer, J., et al., "LUSH--A Computer Program for Complex Response Analysis of Soil-Structure Systems", Berkeley, Univ. of Calif., Earthquake Engineering Research Center, Report No. EERC 74-4, 1974.

3.7.2-6 Schnabel, P.B., J. Lysmer, and H. B. Seed, "SHAKE--A Computer Program for Earthquake Response Analysis of Horizontal Layered Sites", Berkeley, Univ. of Calif., Earthquake Engineering Research Center, Report No. EERC 72-12, 1972, pp. 88.

3.7.2-7 Houston Lighting & Power Company, May 19, 1976, letter to Nuclear Regulatory Commission, Correspondence No. ST-H1-AE-152.

3.7.2-8 Lysmer, J., T. Udaka, C. F. Tsai, and H. B. Seed, "FLUSH, a Computer Program for Approximate 3-D Analysis of Soil-Structure Interaction Problems", Berkeley, Univ. of Calif., Earthquake Engineering Research Center, Report No. EERC 75-30 (November, 1975).

REFERENCES (Continued)

3.7.2-9 ASCE, "Structural Analysis and Design of Nuclear Plant Facilities", draft for trail use and comment (supplement), prepared by the Editing Board and Task Groups of the Committee on Nuclear Structures and Materials, Structural Division, American Society of Civil Engineers (July 1976).

3.7.2-10 Udaka, T., "Analyses of Response of Large Embankments to Traveling Base Motions", Ph.D. dissertation, Univ. of Calif. (Berkeley, 1975).

3.7.2-11 "Soil-Structure Interaction Analysis Re-evaluation For South Texas Project", Bechtel Report No. S-001.

3.7.2-12 "Soil-Structure Interaction Studies for South Texas Project", Report prepared by Woodward-Clyde Consultants.

3.7.2-13 "Computations of Spring and Damping Coefficients for Category I Structures, South Texas Project, Units 1 and 2", Woodward-Clyde Consultants, April 1980.

3.7.2-14 Ibrahim, A.M. and Hadjian, A.H., "The Composite Damping Matrix for a Three Dimensional Soil-Structure System", 2nd ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities, pp. 932, New Orleans, 1975.

3.7.2-15 Pajuhesh, J. and Hadjion, A.H., "Dynamic Interaction of Components, Structures and Foundations of Nuclear Power Facilities", Transactions of 4th SMIRT Conference, San Francisco, California, Paper K 3/9, 1977.

3.7.3-1 "Newmark, N. M., J. A. Blume and K. K. Kapur, "Design Response Spectra for Nuclear Power Plants", ASCE Structural Engineering Meeting, San Francisco, April 1973.

3.7.3-2 "Damping Values of Nuclear Plant Components", WCAP-7921- AR, May 1974.

3.7.3-3 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.

3.7.3-4 Biggs, J. M., Introduction to Structural Dynamics, McGraw-Hill, (New York, 1964).

3.7.3-5 Thomas, T. H., et al., "Nuclear Reactors and Earthquakes", TID-7024, U. S. Atomic Energy Commission, Washington, D.C., August 1963.

REFERENCES (Continued)

3.7.3-6 Chan, S. P., H. L. Cox, and W. A. Benfield, "Transient Analysis of Forced Vibration of Complex Structural- Mechanical System", J. Royal Aeronautical Society, July 1962.

3.7.3-7 Gesinski, T. L., and D. Chiang, "Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and loss of Coolant Accident", WCAP-8236, December 1973 (Proprietary) and WCAP-8288, January 1974 (Nonproprietary).

3.7.3-8 Gesinski, T. L., "Fuel Assembly Safety Analysis for Combined Seismic and loss of Coolant Accident", WCAP-7950, July 1972.

3.7.3.9 "Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident", WCAP-8236, Addendum 1, March 1974 (Proprietary) and WCAP-8288, Addendum 1, April 1974 (Nonproprietary).

3.7.3-10 Yeh, G. K., "Seismic Analysis of Slender Buried Beams", Seismological Society of America Bulletin, Vol. 64, No. 5 (October 1974), pp 1551-1562.

3.7.3-11 Shah, H. H., and S. L. Chu "Seismic Analysis of Underground Structure Elements", Journal of Power Division of ASCE, Vol. 100, No. P01 (July 1974), pp 53-62.

TABLE 3.7-1

DAMPING VALUES[1]

(Percent of Critical Damping)

Structure or Component	Operating Basis Earthquake[2]	Safe Shutdown Earthquake
Equipment and large-diameter piping systems,[3] pipe diameter greater than 12 in.	2	3
Small-diameter piping system, 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

Table 3.7-1 is derived from the recommendations given in Reference 3.7.3-1 and complies with RG 1.61, October 1973.

1. These damping values are for non-NSSS equipment. See Table 3.7-7 for damping values of NSSS equipment.

2. In the dynamic analysis of active components as defined in RG 1.48, these values should also be used for SSE.

3. Includes both material and structural damping. If the piping system consists of only one or two spans with little structural damping, use values for small-diameter piping.

TABLE 3.7-2

METHOD OF SEISMIC ANALYSIS

USED FOR CATEGORY I STRUCTURES

Structures		Method of Analysis (Notes): Modal Response Spectrum	Method of Analysis (Notes): Modal Time-History
Reactor Containment Building (RCB)			• (1) (6) (7)
Mechanical-Electrical Auxiliary Building (MEAB)			• (1) (2)
Fuel Handling Building (FHB)			• (1) (7) (8)
Diesel Generator Building (DGB)			• (1) (7) (9)
Essential Cooling Water Intake Structure			• (3)
Essential Cooling Water Discharge Structure		• (4)	
Auxiliary Feedwater Storage Tank (AFST)		• (10)	
Underground Piping and Electrical Raceway System	(5)	N.A.	N.A.
Cable Tray Supports and HVAC Duct Supports		• (11)	

1. Two-step finite element method (FEM) for soil-structure interaction (SSI) analysis is used (design-basis acceleration response spectra).

2. Two-step FEM for SSI analysis, incorporating revised building configuration (configuration-revised spectra), is used only for the seismic design of piping and pipe supports in the MEAB. The design-basis spectra as well as the configuration-revised spectra are conservative with respect to the confirmatory-basis spectra defined in Section 3.7.2.4.

3. Elastic half-space (EHS) method for SSI analysis is used.

4. Equivalent Static Method is used for structural design based on free-field peak spectral accelerations amplified by a factor of 1.5.

TABLE 3.7-2 (Continued)

METHOD OF SEISMIC ANALYSIS
USED FOR CATEGORY I STRUCTURES

Notes (Continued):

5. Wave propagation method is used, as described in Section 3.7.3A.12.

6. The design-basis response spectra at nozzle attachment points within the Westinghouse (W) reactor coolant system are developed by enveloping the EHS and the two-step FEM solutions obtained through mathematical models that incorporate the W NSSS Seismic model coupled to the RCB structural model.

   Subsequent verification of the design-basis nozzle point response spectra to account for changes in the NSSS support stiffnesses and updated linearization of the model, is performed by the EHS method.

7. The EHS solution results in horizontal spectral response augmented in the low frequency range. The affected design-basis spectra are annotated to assure incorporation of the EHS-augmented spectra.

8. The analysis of the FHB along the E-W direction is based on a fixed-base model excited with the free-field ground motion amplified by a factor of 1.4 to account for SSI. (Original analyses based on two-step FEM for SSI did not include the E-W direction for the FHB.)

9. The analyses of the DGB along the E-W direction and the vertical direction are based on a fixed-base model excited with the free-field ground motions amplified by a factor of 1.4 to account for SSI. (Original analyses based on two-step FEM for SSI did not include the E-W and vertical directions for the DGB.)

10. Structural Design of the AFST along horizontal direction is based on a fixed-base model excited with input motion defined by the RG 1.60 design spectra. Acceleration response spectra used for verification of piping seismic design were developed by EHS method.

11. Equivalent static method is used with design accelerations equal to 1.0 times the peak spectral acceleration from applicable floor response spectra. Dynamic analyses of representative models of these subsystems are performed to demonstrate that the equivalent static method with a factor of 1.0 is justified by the calculated dynamic response.

TABLE 3.7-3

REACTOR CONTAINMENT BUILDING

NATURAL FREQUENCIES*

Mode No.	Frequency (CPS)	Mode No.	Frequency (CPS)
1	1.53	19	14.03
2	1.53	20	14.30
3	3.13	21	14.33
4	3.45	22	16.38
5	3.46	23	16.75
6	3.48	24	19.16
7	6.00	25	19.63
8	6.35	26	20.18
9	8.16	27	23.11
10	8.99	28	23.15
11	9.24	29	24.54
12	12.35	30	25.04
13	12.40	31	25.41
14	12.71	32	25.91
15	12.85	33	26.29
16	13.07	34	26.56
17	13.41	35	27.77
18	13.45	36	29.54

* These natural frequencies are obtained from the EHS method for SSI Analysis.

TABLE 3.7-4

MECHANICAL-ELECTRICAL AUXILIARIES BUILDING

NATURAL FREQUENCIES*

Mode No.	Frequency (CPS)	Mode No.	Frequency (CPS)
1	1.88	11	19.06
2	1.89	12	23.49
3	2.16	13	24.54
4	2.17	14	25.10
5	2.58	15	26.25
6	2.81	16	27.73
7	9.44		
8	11.95		
9	14.04		
10	17.29		

* These natural frequencies are obtained from the EHS method for SSI Analysis.

TABLE 3.7-5

DIESEL GENERATOR BUILDING

NATURAL FREQUENCIES*

Mode No.	Frequencies (CPS)
1	2.66
2	2.74
3	4.56
4	5.32
5	5.97
6	6.20
7	16.20
8	22.97
9	24.20
10	26.46
11	27.10
12	30.26
13	31.36
14	32.68

* These natural frequencies are obtained from the EHS method for SSI Analysis.

TABLE 3.7-6

FUEL HANDLING BUILDING

NATURAL FREQUENCIES*

Mode No.	Frequency (CPS)	Mode No.	Frequency (CPS)
1	0.22 (Convective Mode)	15	9.68
2	0.37 (Convective Mode)	16	9.69
3	1.68	17	9.85
4	1.77	18	10.63
5	1.99	19	12.21
6	2.23	20	15.19
7	2.48	21	18.47
8	3.27	22	19.35
9	3.43	23	19.54
10	3.49	24	19.61
11	4.47	25	21.04
12	4.78	26	23.36
13	5.13	27	27.25
14	6.16	28	30.53

* These natural frequencies are obtained from the EHS method for SSI Analysis.

TABLE 3.7-7

DAMPING VALUES USED FOR SEISMIC ANALYSIS OF NSSS EQUIPMENT

Item	Damping (Percent Critical) Upset Conditions (OBE)	Damping (Percent Critical) Faulted Conditions (SSE, DBA)
Primary Coolant Loop System - components and large piping (applicable to 12-in.-diameter or larger piping)	2	4
Small piping	1	2
Welded steel structures	2	4
Bolted and/or riveted steel structures	4	7

TABLE 3.7-8

CASES WHERE THE CONFIRMATORY-BASIS SPECTRA ARE USED FOR THE JUSTIFICATION OF COMPLETED SEISMIC ANALYSIS, DESIGNS AND/OR QUALIFICATIONS

1.	Seismic qualifications of a battery rack.	The confirmatory basis spectra at EL. 35 ft. of the MEAB were used for the seismic qualification of an existing design of a Class 1E, 125 VDC battery rack (Battery No. NCX 1200).
2.	Verification of the DGB Foundation Mat Design	The design-basis seismic analysis of the Diesel Generator Building (DGB) in the E-W direction was originally based on fixed-base mathematical models excited with the free-field ground motion amplified by a factor of 1.4 to account for SSI (See Table 3.7-2). This fixed-base analysis was a rudimentary approximation for the SSI, and resulted in E-W acceleration responses that are very conservative with respect to both the design-basis two-step FEB solution which was performed only along the N-S direction, and the confirmatory-basis solutions along the E-W and N-S directions. Subsequently, the zero-period accelerations (ZPA's) of the conservative design-basis analysis along the E-W direction were reduced by 50% in order to reconcile the existing design of the foundation mat for the DGB. The reduced value adopted for the E-W ZPA is justified by (1) being of the same order of magnitude as the N-S ZPA determined from the design-basis analysis by two-step FEM, and (2) being at least 20% higher than the ZPA obtained from the E-W confirmatory-basis spectra. Therefore, the confirmatory-basis analyses were used only as part of the justification for the lower E-W ZPA used in the reconciliation of the DGB foundation mat design, and have not been used in any other way for structural design or seismic qualification in the DGB.
3.	Seismic qualifications of the MEAB 480V Motor Control Centers (MCCs)	The confirmatory-basis spectra at El. 60 ft. of the MEAB are being used for the seismic qualification of an existing design of the Class 1E, 480V MCC (3E171EMCE1C1, 1C2, 1C4).

TABLE 3.7-9

MAXIMUM RELATIVE DISPLACEMENTS

AMONG PRINCIPAL POWER BLOCK STRUCTURES

	Max. Relative Displacement (in.)		
Interface	OBE	SSE	Long Term Diff. Settlement*
RCB/MEAB	0.14	0.23	1.0
RCB/FHB	0.22**	0.44**	1.0
DGB/MEAB	0.03	0.06	1.0
FHB/MEAB	0.16	0.28	1.0

* These values represent the established design criteria for differential movement. The values are derived from differential settlement projections, and are subject to ongoing monitoring to assure consistency with the periodically measured settlements of controlled locations. To date, the actual, measured settlements agree with the predicted settlements. The differential settlement criteria is discussed at length in Section 2.5.4.11 and the predicted and actual differential settlement values are reported in Appendix 2.5.C.

** These design-basis values are slightly lower than those obtained from the single-step finite element seismic analysis by "FLUSH" at building elevations higher than 52 ft. However, it has been determined that this slight discrepancy does not affect the seismic design of interconnecting piping anchored in the building of FHB and RCB.

APPENDIX 3.7.A

DYNAMIC LATERAL EARTH PRESSURES

3.7.A-1 Introduction

Dynamic lateral earth pressures were used in the analysis of major Category I structures. Two methods have been used to determine the dynamic lateral earth pressures, namely pseudostatic method (Ref. 3.7.A-1) and finite element, soil/structure interaction (SSI) method (Section 3.7.A.2). Comparison of results from both methods indicates:

1. For building walls not subject to surcharge loadings, the pseudostatic method and the SSI analysis gave approximately the same dynamic lateral earth pressure.

2. For building walls subject to surcharge loading connecting to top of the walls (i.e., walls of the Fuel-Handling Building [FHB] surcharge by higher floors of the same building and tendon gallery walls beneath the Reactor Containment Building [RCB]), the SSI analysis resulted in lower lateral earth pressures as compared to the pseudostatic method.

3. For building walls subject to surcharge loadings from immediately adjacent, structurally separate buildings (i.e., east wall of the RCB adjacent to the Auxiliary Building and north wall of the Auxiliary Building adjacent to the Diesel-Generator Building [DGB]) the SSI analysis resulted in greater lateral earth pressures.

   The pseudostatic lateral earth pressures were used in the preliminary analyses of the Category I structures. For the cases discussed in item 1, above, where the pseudostatic and SSI methods resulted in approximately the same pressures, the final design was based on the pseudostatic pressures. For the cases discussed in item 2, above, where the pseudostatic method resulted in higher lateral earth pressures, the pseudostatic pressures were used in the design to provide a conservative analysis. For the cases discussed in item 3, above, where the SSI analysis resulted in higher pressures, the SSI results were used in the design of structures.

3.7.A.2 General Procedures

Dynamic lateral earth pressures on the walls of structures and base shear forces at the foundations of structures were calculated from the dynamic finite element SSI analyses that are presented in Section 3.7.

The SSI analyses were performed using the computer code WCC*LUSH 4 (Ref. 3.7.A-2). An auxiliary computer code, WCC*FORCE 2 (Ref. 3.7.A-2), was used to obtain dynamic force time-histories at the nodal points at soil/structure interfaces. The results from the FORCE 2 program were used in obtaining all the dynamic pressures and forces presented herein except on the inside walls of the tendon galleries beneath the RCB. In this location, pressures were obtained from stresses in the adjacent elements using the computer code WCC*STRESS 2 (Ref. 3.7.A-2).

The forces and pressures computed from the SSI analyses and presented herein represent those values which occur at the instant of time at which the total dynamic force on a given wall or foundation is a maximum value. As described in Section 3.7, separate SSI analyses were made for the horizontal and vertical components of input excitation. The resultant pressures and forces due to horizontal and vertical components of the input were then combined as the square root of the sum of the squares (SRSS). For example, for a given wall and for the case of the horizontal component of the Operating Basis Earthquake (OBE), a horizontal force and pressure distribution on the wall were calculated corresponding to a time when total dynamic force on the wall was a maximum value. For the case of the vertical component of the OBE, a horizontal force and pressure distribution on the wall were calculated in the same manner. The resulting two sets of pressures and forces on the wall were then combined as the SRSS.

The SSI analysis consisted of OBE and Safe Shutdown Earthquake (SSE) design earthquakes for cross section 1 and OBE design earthquake for both cross sections 2 and 3. As indicated in the analysis results of the following section, the lateral earth pressures for cross section 1 due to the SSE are less than twice the pressures calculated for the OBE. Therefore, the analyses for cross sections 2 and 3 were performed for the OBE only, and the lateral earth pressures due to the SSE for these cross sections were assumed to be not more than twice the OBE pressures. Table 3.7.A-1 presents the directions of earthquake excitation and dynamic soil properties considered in these analyses. The locations of the analysis cross sections and the cases are described in detail in Section 3.7.

3.7.A.3 Dynamic Earth Pressures, Cross Section1

The finite element model of cross section 1 is shown on Figure 3.7.A-1. Calculated dynamic earth pressures for the case of average soil properties are shown on Figures 3.7.A-2 and 3.7.A-3 (east side of the Auxiliary Building for the OBE and SSE, respectively), Figures 3.7.A-4 and 3.7.A-5 (east side of the RCB adjacent to the Auxiliary Building), and Figures 3.7.A-6 and 3.7.A-7 (west side of the RCB).

Maximum dynamic earth pressures were also calculated on the inside and outside walls of the tendon gallery beneath the RCB. The results of these calculations are summarized in Table 3.7.A-2.

The effects of variation in soil properties on the dynamic earth pressures are indicated on Figures 3.7.A-2 and 3.7.A-7 and in Table 3.7.A-2.

3.7.A.4 Dynamic Earth Pressures, Cross Section 2

The finite element model of cross section 2 was made for the OBE and is shown on Figure 3.7.A-8. The dynamic earth pressure distributions obtained on the upper and lower walls of the FHB for average soil properties are shown on Figures 3.7.A-9 and 3.7.A-10. The pressure distribution on the north wall of the RCB is shown on Figure 3.7.A-11. Maximum dynamic earth pressures on the tendon gallery walls on the north side of the RCB are summarized in Table 3.7.A-2. Dynamic earth pressures were also calculated on the tendon gallery walls on the south side of the RCB, adjacent to the FHB. However, since the foundation elevation of the lower level of the FHB is at the same elevation as the base of the adjacent tendon gallery, dynamic earth pressures will not be transmitted to the portion of the wall of the tendon gallery which is immediately adjacent to the FHB.

The effects of soil property variations on the dynamic earth pressures are indicated on Figures 3.7.A-9 through 3.7.A-11 and in Table 3.7.A-2.

3.7.A.5 Dynamic Earth Pressure, Cross Section 3

The finite element model of cross section 3 is shown on Figure 3.7.A-12. The SI analysis was conducted for the horizontal component of the OBE for average soil properties. The computed dynamic earth pressures are shown on Figure 3.7.A-13 for the north wall of the Auxiliary Building adjacent to the DGB and on Figure 3.7.A-14 for the south wall of the Auxiliary Building.

The contribution of the vertical component to the dynamic pressures shown on Figures 3.7.A-13 and 3.7.A-14 has been estimated as discussed hereunder. For the north wall of the Auxiliary Building adjacent to the DGB (Figure 3.7.A-13), it was assumed that the distribution of dynamic pressure due to the vertical component would be similar to the distribution obtained for the vertical component on the east wall of the RCB, cross section 1, (Figure 3.7.A-4) and in proportion to the gross-bearing pressures of the adjacent structures. Thus, the dynamic pressure distribution due to the vertical component shown on Figure 3.7.A-13 was obtained by multiplying the dynamic pressure distribution due to the vertical component shown on Figure 3.7.A-4 by the ratio of the gross-bearing pressure of the DGB, section 3, to the gross bearing pressure of the Auxiliary Building, section 1. This approach is reasonable because the presence of a structure adjacent to a wall has been found to significantly influence the dynamic pressures on the wall and because, as shown on Figure 3.7.A-4, the vertical component contributes only slightly to the total dynamic pressure on the wall. Thus, any reasonable estimating procedure for the vertical component will not significantly affect the total pressure on the wall.

For the south wall of the Auxiliary Building in cross section 3 (Figure 3.7.A-14), the dynamic pressure due to the vertical component of the OBE was taken to be the same as calculated for the east wall of the same building in cross section 1 (Figure 3.7.A-2). Based on a comparison of the results for the horizontal component of the OBE on Figures 3.7.A-2 and 3.7.A-14, this procedure is conservative.

Based on the results for cross section 1, the soil property variations have a small effect on dynamic earth pressures in cross section 3; the estimated effect is indicated on Figures 3.7.A-13 and 3.7.A-14.

3.7.A.6 Base Shear Forces

The maximum total base shear forces acting on the buildings at an instant of time are summarized in Table 3.7.A-3. The values represent the SRSS of the maximum horizontal base shear forces calculated for the horizontal and vertical components of the input motion. As would be expected, it was found that the vertical component of the OBE or SSE contributed relatively little to the base shear; the resultant SRSS values shown in Table 3.7.A-3 exceed those due to the horizontal component alone by amounts varying from approximately 0 to 7 percent.

For cross section 2 and 3, the base shear forces due to the SSE have been assumed equal to twice the forces calculated for the OBE for these cross sections. The effects of soil property variations on base shear forces are summarized in Table 3.7.A-3. In most cases, the analyses using average soil properties resulted in the highest base shear forces.

REFERENCES

Appendix 3.7.A:

3.7.A-1 Seed, H. B. and R. V. Whitman, "Design of Earth-Retaining Structures for Dynamic Loads", ASCE Specialty Conference on Lateral Stresses in the Ground and Design of Earth-Retaining Structures, 1970.

3.7.A-2 Woodward-Clyde Consultants, WCC Computer Program Documentation File.

TABLE 3.7.A-1

FINAL ANALYSIS CASES

SOIL/STRUCTURE INTERACTION STUDIES

Cross-Section	Design Earthquake	Direction of Excitation	Dynamic Soil Properties		
			Average	Upper Bound	Lower Bound
1	OBE	Horizontal	X	X	X
		Vertical	X		
	SSE	Horizontal	X		
		Vertical	X		
2	OBE	Horizontal	X	X	X
		Vertical	X		
3	OBE	Horizontal	X		

TABLE 3.7.A-2

CALCULATED DYNAMIC PRESSURES ON TENDON GALLERIES

Cross Section	Tendon Gallery Walls		Maximum Wall Pressure (lb/ft^2) OBE	During SSE
1	West Side	Outside Wall	330	610
		Inside Wall	390	750
	East Side	Outside Wall	360	660
		Inside Wall	400	760
2	North Side	Outside Wall	490	980
		Inside Wall	380	760

1. For cross section 2, dynamic pressures during SSE are assumed equal to twice the OBE values.

2. Values in table are for average soil properties. Analyses for average properties resulted in highest pressures except for cross section 2, upper-bound properties would increase pressures on outside wall, north side by approximately 3 percent.

TABLE 3.7.A-3

CALCULATED BASE SHEAR FORCES ON BUILDINGS

Cross Section	Building	Direction of Base Shear on Building	Base Shear Force (kips) During OBE	Base Shear Force (kips) During SSE
1	Reactor Building	East	72	130
		West	69	134
	Auxiliary Building	East	37	75
		West	38	73
2	Reactor Building	North	67	134
		South	80	160
	Fuel-Handling Building	North	46	92
		South	47	94
3	Auxiliary Building	North	61	122
		South	58	116
	Diesel-Generator Building	North	13	26
		South	12	24

1. Base shear forces are for a 1-ft width perpendicular to the analysis cross section.

2. For cross sections 2 and 3, base shear forces during the SSE are assumed equal to twice the OBE values.

3. For cross section 3, the contribution of the vertical component of input motion to the base shear forces was estimated to be negligible based on the results for the Auxiliary Building in cross section 1.

4. Values in table are for average soil properties. Analyses for average properties resulted in highest base shear forces except: for cross section 1, upper-bound properties would increase the base shear forces on the Reactor Building by approximately 12 percent; for cross section 2, upper-bound properties would increase the base shear forces on the FHB by approximately 6 percent.

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 Concrete Containment

3.8.1.1 Description of the Containment.

3.8.1.1.1 General Description: The Containment is a fully continuous, steel-lined, post-tensioned, reinforced-concrete structure consisting of a vertical cylinder with a hemispherical dome, supported on a flat foundation mat. The cylinder and dome are post-tensioned with high-strength unbonded wire tendons. The dimensions of the Containment are: 150 ft inside diameter, 239-1/4 ft inside height to the top of the dome, with 4 ft cylinder wall thickness, 3 ft dome thickness, and 18-ft mat thickness. The top of the foundation mat is 41-1/4 ft below grade.

A continuous, reinforced-concrete tendon gallery is located at the perimeter of the mat with floor of the gallery extending 5-1/2 ft below the base of the mat. The gallery is 7-2/3 ft wide and 11 ft high and is provided for the installation and surveillance of the vertical post-tensioning system. The bottom of the tendon gallery is 67-1/4 ft below grade. Access to the tendon gallery is provided by a shaft from the ground level to the tendon gallery. Emergency access to the gallery is provided through the Mechanical-Electrical Auxiliaries Building (MEAB).

The Containment wall is independent of the adjacent interior and exterior structures, with sufficient space being provided between the Containment wall and the adjacent structures to prevent contact under all loading conditions.

The Containment encloses the reactor vessel, pressurizer, steam generators (SGs), reactor coolant pumps (RCPs), and loops and portions of the Auxiliary and Engineered Safety Features (ESF) Systems.

The Containment is designed such that during accident conditions, water introduced into the Containment will not flood the cavity below the reactor vessel to the extent that the water will contact the bottom of the reactor vessel while it is hot and pressurized before the contents of the refueling water storage tank (RWST) have been injected.

The Containment protects the Reactor Coolant System (RCS) from site environmental conditions. It is designed as a Category I structure for earthquake, tornado and external missile-loading conditions.

The Containment also limits the release of radioactive fission products to the environment in the unlikely event of a Loss-of-Coolant Accident (LOCA), in addition to providing biological shielding for both normal and accident conditions.

For Containment drawings, refer to Figures 3.8.1-1 through 3.8.1-7.

3.8.1.1.2 Foundation Mat: The foundation mat is a conventionally reinforced concrete mat of circular shape and uniform thickness. Reinforcement is placed in a rectangular grid in combination with radial and hoop bars on both the top and bottom faces of the mat. A continuous tendon gallery is provided below the mat at its periphery for installation and inspection of vertical tendons (see Figure 3.8.1-2 for details).

3.8.1.1.3 Steel Liner: A continuous welded steel liner plate is provided on the entire inside face of the Containment to limit the release of radioactive materials into the environment. The nominal thickness of the liner in the wall and dome is 3/8 inch. A 3/8-inch-thick plate is used on top of the foundation mat and is covered with a 24 in. concrete fill slab.

An increased plate thickness up to 2 in. is provided around all penetrations and for the crane girder brackets.

An anchorage system is provided to prevent instability of the liner. For the dome, the anchorage system consists of meridional structural tees, circumferential angles, and plates, while for the cylinder, a system of vertical and circumferential stiffeners, using structural angles, channels, and plates, is provided.

Leak chase channels and angles are provided at the bottom liner seams which, after construction, are inaccessible for leaktightness examination due to the 2-ft interior fill slab.

For typical liner details, see Figure 3.8.1-6.

3.8.1.1.4 Arrangement of Shell Reinforcement: The cylindrical wall is reinforced with conventional steel reinforcing bars throughout the structure. The bars are placed in a horizontal and vertical pattern in each face of the cylinder wall. Additional bars are provided around penetrations and in the buttresses to resist local stress concentrations. Radial shear reinforcement is provided throughout, and tangential shear reinforcement is provided where required.

The reinforcement in the dome is provided in a meridional and circumferential pattern up to 45 degrees from the spring line, with the remaining area being reinforced using a grid pattern. Reinforcement is provided on both faces of the dome wall. Radial ties are provided to both resist radial shear and prevent delamination of the dome under prestressing.

For details of the reinforcement arrangement, see Figures 3.8.1-2 and 3.8.1-3.

3.8.1.1.5 Arrangement of Post-Tensioning Tendons: The cylindrical portion and the hemispherical dome of the Containment are prestressed by a post-tensioning system consisting of horizontal and vertical tendons. Three buttresses are equally spaced at 120 degrees around the Containment.

The cylinder and the lower half of the dome are prestressed by horizontal tendons anchored 360 degrees apart, bypassing the intermediate buttresses. Each successive hoop tendon is progressively offset 120 degrees from the one beneath it. The vertical U-shaped tendons are continuous over the dome, forming a two-way system for the dome. These tendons are anchored in the continuous gallery beneath the base mat.

The tendons are placed in embedded-tendon sheaths which are filled with a corrosion inhibitor. For tendon arrangement, see Figures 3.8.1-1 and 3.8.1-7.

3.8.1.1.6 Containment Penetrations and Attachments: Access into the Reactor Containment Building (RCB) is provided by an equipment hatch, a personnel airlock, and an auxiliary airlock. The equipment hatch is a 24-foot inside diameter, single-closure penetration, as shown on Figure 3.8.1-4. It consists of a welded steel barrel furnished with a double O-ring gasket and a bolted, dished door. The personnel airlock is an 11-foot-6-inch inside diameter, welded-steel assembly with double doors. The auxiliary airlock is a 5-foot-5-inch inside diameter, welded-steel assembly with double doors.

Other penetrations through the Containment include the electrical penetrations, the piping penetrations, and the fuel transfer tube. All penetrations are pressure-resistant, leaktight, welded assemblies. The penetration sleeves are welded to the liner and anchored into the concrete Containment wall. For typical details, see Figures 3.8.1-8 through 3.8.1-12.

The fuel transfer tube penetration between the refueling canal in the RCB and the spent fuel pool in the Fuel Handling Building (FHB) consists of a stainless steel pipe inside a carbon steel sleeve. The inner pipe acts as a transfer tube; the outer tube is welded to the Containment liner. Bellows expansion joints are provided to permit differential movement. For typical details, see Figure 3.8.1-8.

Canister-type penetrations are used for electrical conductors passing through the Containment. The penetration canisters are installed in steel penetration sleeves welded into the wall of the Containment liner. Sealing between the canisters and the sleeves is accomplished by welding. For typical details, see Figure 3.8.1-12.

Piping penetration assemblies are generally of three types, the type of penetration used for a particular line being dependent on the service requirements of that line. A high-energy penetration is used where the temperature or pressure of the fluid is high and considerable thermal movement of the line can be expected. Moderate-energy penetrations are used where little or no thermal movement of the process line is anticipated. Multiple penetrations are used where more than one pipe goes through a penetration. For typical details, see Figures 3.8.1-10 and 3.8.1-11.

The crane girder support brackets are welded to a section of the liner plate and anchored into the Containment concrete wall, as shown on Figure 3.8.1-6.

Typical joint details at liner plate and reinforcing steel cadwelds for connection of shield walls to base mat are shown on Figure 3.8.3-3.

3.8.1.2 Applicable Codes, Standards and Specifications.

3.8.1.2.1 Design Codes: The basic code used in the design of the Containment is the "Proposed Standard Code for Concrete Reactor Vessels and Containments," American Concrete Institute (ACI) 359 - American Society of Mechanical Engineers (ASME) Code, Section III, Division 2, issued for trial use and comment in 1973, including subsequent addenda 1 through 6. Herein-after, this code shall be referred to as the ASME-ACI 359 document. Exceptions to the code are as follows:

- Authorization and stamping requirements in Subsection CA
- Personnel qualifications for Level III Inspection Engineer
- The filing and certification of those design and construction documents required by Subsections CA-3200 and CA-3300, which are required only for stamping (The information required by these subsections will be available, but not necessarily in the format specified)
- The exception described in Section 3.8.1.6.3

Additional codes used in the design of the Containment are:

1. American Society of Mechanical Engineers - ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NE for Class MC components, 1971, including the Winter 1973 addenda; ASME B&PV Code, Section IX and Section II, 1971 including the Winter 1973 addenda
2. American Institute of Steel Construction - AISC Specification for the Design Fabrication and Erection of Structural Steel for Buildings, 1969, including supplements 1, 2, and 3
3. American National Standards Institute - ANSI A58.1-1972, "American Standard Building Code Requirements for Minimum Design Loads in Buildings and Other Structures"

3.8.1.2.2 Government Regulations and Regulatory Guides: The design, construction, materials, testing, examination, etc., of the Containment are in conformance with government regulations as discussed in Section 3.1 and with the following NRC Regulatory Guides (RGs) as noted in Section 3.12.

RG 1.10 "Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures", Revision 1

RG 1.15 "Testing of Reinforcing Bars for Category I Concrete Structures", Revision 1

RG 1.18 "Structural Acceptance Test for Concrete Primary Reactor Containments", Revision 1

RG 1.19 "Nondestructive Examination of Primary Containment Liner Welds", Revision 1

RG 1.35 "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures", Proposed Revision 3

RG 1.55 "Concrete Placement in Category I Structures", Revision 0

RG 1.57 "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components", Revision 0

RG 1.69 "Concrete Radiation Shields for Nuclear Power Plants", Revision 0

RG 1.76 "Design Basis Tornado for Nuclear Power Plants", Revision 0

The following guides are not applicable to South Texas Project Electric Generating Station (STPEGS) per the implementation portion of the guide; however, degree of compliance is addressed in the Updated Final Safety Analysis Report (UFSAR).

RG 1.94 "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants", Revision 1

RG 1.103 "Post-Tensioned Prestressing Systems for Concrete Reactor Vessels and Containments", Revision 1

An exception is taken to RG 1.10, "Mechanical (Cadweld) Splices in Reinforcing Bars". For further explanation of this position, see Section 3.8.1.6.3. Exceptions are taken to RG 1.35 as discussed in Section 3.8.1.7.3.

3.8.1.2.3 Specifications and Standards: The specifications and standards are used as a basis for the construction, inspection, materials, and testing of the Containment structure.

1. American Society for Testing and Materials - ASTM Standards as referenced in the ASME-ACI 359 document and Section III of the ASME B&PV Code. Different issue dates of ASTM standards may be used provided they meet the minimum technical requirements as stated herein.

2. American Concrete Institute - ACI Manual of Standard Practice

3. Prestress Concrete Institute (PCI) - "Tentative Specification for Post-Tensioning Materials", as reported by the PCI Post-Tensioning Subcommittee, PCI Journal (January - February 1971)

4. American Institute of Steel Construction - AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings", 1969, including supplements 1, 2, and 3.

5. American Welding Society (AWS) - AWS D1.1-75, "Structural Welding Code". Visual inspection acceptance criteria for welding in conformance with AWS D1.1 are specifically defined in Appendix 3.8.B. The criteria are incorporated in construction specifications where field welding per AWS D1.1 is specified. The polar crane runway girder welding is in accordance with AWS D1.1 (1972) including revision through 1974.

6. American National Standards Institute - ANSI N45.4-1972, "Leakage Rate Testing of Containment Structure for Nuclear Reactors"

7. Army Corps of Engineers (C of E) - CRD C39, "Coefficient of Thermal Expansion", and C44, "Coefficient of Thermal Conductivity"; CRD-C621, Standard Specification for Packaged Dry, Hydraulic-Cement Grout (Nonshrink)

3.8.1.3 Loads and Loading Combinations.

3.8.1.3.1 Definitions of Loads: The following nomenclature and definitions apply to all the loads to be encountered and/or to be postulated in the design of the Containment.

1. Dead Loads (D)

Dead load of the structure plus any other superimposed permanent loads, except prestressing forces. Included are the weights and operating loads of specific major equipment as specified by the equipment manufacturers. Hydrostatic loads and crane loads are also treated as dead load.

The polar crane bridge's rated lift capacity is 417 tons (Unit 1)/500 tons (Unit 2). Runway girders and supporting brackets are designed to the highest lift capacity (500 tons).

Hydrostatic loads are calculated assuming the water table at El. 27 ft and a unit weight of water at 62.4 lb/ft3. A reinforced concrete density of 145 lb/ft3 is used in the calculation of dead load.

2. Live Loads (L)

Floor live loads which account for movable loads and maintenance loads. Also considered are the construction loads, lateral soil pressure loads and a minimum roof load of 12 lb/ft2 on the dome.

Horizontal and vertical impact loads are considered in accordance with the AISC Specification.

Lateral soil pressure loads including pressures resulting from adjacent foundation loads are calculated as indicated in Section 2.5.4.10.5.

3. Prestressing Loads (F)

The prestressing load to be considered is the initial prestressing load, F_i, which occurs when the prestressing tendons are subjected to the most critical stress during the initial tensioning, and the effective prestressing load, F_e, which considers the time-dependent losses for the life of the plant.

The initial prestress load, F_i, is calculated based on a tendon ultimate strength of 240 kip/in.2 with initial jacking of tendon to 80 percent ultimate and lockoff stress of 70 percent ultimate. Effective prestress load, F_e, includes long-term prestress losses of 14.1 percent in the vertical and 15.8 percent in the hoop tendons.

The average effective prestressing forces, including the effect of surveillance tendons, used in the Containment analysis are as follows.

For hoop tendon:

a. An external pressure of 6,872 lb/ft2 based on a 532 kip/ft hoop stress resultant from El. (-)5 ft-6 in. to 9 ft-6 in.

b. An external pressure of 9,946 lb/ft2 based on a 770 kip/ft hoop stress resultant from El. 9 ft-6 in. to 153 ft

c. An external pressure of 8,890 lb/ft2 based on a 684 kip/ft hoop stress resultant from El. 153 ft to 10 degrees on the dome

d. An external pressure of 6,159 lb/ft2 based on a 415 kip/ft hoop stress resultant from 10 to 45 degrees on the dome

For vertical dome tendons:

The vertical dome tendons produce an external pressure of approximately 5,465 lb/ft². This pressure varies over the surface of the dome.

4. Design Basis Accident (DBA) Pressure Loads (P_a)

The minimum equivalent static design pressure (P_a = 56.5 psig) is chosen conservatively above the peak pressure occurring as a result of a DBA (see Section 6.2 for Containment pressure response analyses).

5. Operating and Shutdown Thermal Loads (T_o)

Operating thermal loads are the most severe thermal conditions for summer and winter operations. Thermal loads are determined on the basis of temperature distributions obtained by heat transfer computations. Reference temperature during construction is assumed to be 60°F. The following temperatures are used in the analysis of the Containment structure:

	Operating Case	Shutdown Case
Summer Operating Thermal Loads (T_{os})		
Containment inside temperature	110°F	65°F
Outside air temperature	95°F	95°F
Soil temperature	75°F	75°F
Winter Operating Thermal Loads (T_{ow})		
Containment inside temperature	110°F	65°F
Outside air temperature	25°F	25°F
Soil temperature	75°F	75°F

6. Test Thermal Loads (T_t)

Thermal loads during pressure test, including liner expansion and temperature gradient in the wall and dome. The summer and winter operating thermal loads (see item 5 above) are applied as the test thermal loads (T_t) in the design of the Containment.

7. Operating Piping Loads (R_o)

Piping thrust and thermal expansion forces and reactions based on the most critical steady-state or transient condition during normal operation or shutdown (Section 3.6).

8. Design Basis Accident Thermal Load (T_a)

Additional thermal effects on structure above normal operating loads, resulting from a DBA.

9. Operating Basis Earthquake (OBE) Loads (E_o)

Loads generated from the OBE. The plant is designed to remain operational under the OBE. The OBE loads are based on a maximum free-field ground acceleration for the site of 0.05g.

In addition to the structural responses, dynamic soil pressures are applied to the structure. The dynamic soil pressures are calculated by the Mononobe-Okabe Method using the same seismic accelerations as used to determine the structural response (Section 3.7).

10. Safe Shutdown Earthquake (SSE) Loads (E_{ss})

Loads generated for the SSE. The structural response and corresponding dynamic soil pressures are determined for the SSE based on a maximum free-field ground acceleration for the site of 0.10g (Section 3.7).

11. Wind Loads (W)

Loads generated by the design basis wind. Wind loads are calculated based on a design wind velocity of 125 mph (Section 3.3). The appropriate pressure coefficients used in calculating the design wind pressure are obtained from American Society of Civil Engineers (ASCE) 3269, "Wind Forces on Structures", for the cylinder and ASCE 4933, "Wind Loads on Dome-Cylinder and Dome-Cone Shapes", for the dome.

12. Tornado Loads (W_t)

Wind, pressure differential and missile loads generated by the design tornado.

The design pressure tornado load is calculated similarly to the wind load using a tornado wind velocity of 360 mph and a gust factor of 1.0 (Section 3.3).

13. External Pressure Load (P_v)

External pressure load of 3.5 psig resulting from pressure variation either inside or outside the Containment.

14. Test Pressure Load (P_t)

The test pressure is equal to 1.15 times the DBA pressure (P_a), in accordance with Section CC-6210 of the ASME-ACI 359 document.

15. DBA Thermal Piping Loads (R_a)

Additional pipe reactions and forces above normal operating loads, due to thermal effects, occurring as a result of a DBA (Section 3.6).

16. Pipe Rupture Loads (Y)

Equivalent static pipe reactions which account for the dynamic effects resulting from a postulated rupture of a high-energy pipe. Also included in this rupture loading are direct jet impingement pressure and missile impact effects due to the postulated break.

17. Flood Loads (H)

Hydrostatic and buoyancy forces due to a failure of the reservoir embankment, additional to the normal hydrostatic forces. Also included are hydrodynamic effects due to wave action. For further details, see Section 3.4.

18. Post-LOCA Flooding

Post-LOCA flooding of the Containment for the purpose of fuel recovery is not a design condition. When access to the Containment is required following a LOCA, all necessary repairs will be made to permit fuel recovery. The layout and design are such that temporary repairs may be accomplished.

3.8.1.3.2 Load Combinations: The design of the Containment incorporates two general loading categories: the Service Load Category and the Nonservice Load Category. Each of these two categories is divided into several conditions of loading, which are further subdivided into several different load combinations as described below.

3.8.1.3.2.1 Service Load Category - This category includes all loading conditions encountered during the construction, test, normal operation, and shutdown periods of the nuclear power plant. The probability of occurrence of these loads is 1.

1. Construction Condition

This condition includes any load applied during construction which would affect the structural integrity and leaktightness of the Containment during its design life span. Loads prior to prestressing, at transfer of prestress and during sustained prestress are considered.

2. Test Condition

This condition includes all loads applied during the structural integrity test.

3. Normal Condition

This condition includes all loads on the structure during normal operation, refueling and shutdown.

4. Severe Environmental Condition

This condition considers all the normal loads on the structure in combination with the loads resulting from an environmental event such as wind or OBE.

A summary of the service load combinations is shown in Table 3.8.1-1.

3.8.1.3.2.2 Nonservice Load Category - This category includes all loading conditions resulting from a system failure and/or those extreme environmental conditions postulated to occur during the life of the plant. Also included in this category is the Severe Environmental Condition. The loads in these conditions occur infrequently in combination with normal operating loads. The design probability of occurrence of some of the infrequent loads, such as the OBE, is one during the life of the plant, while that of other extreme loads, such as tornado and the SSE, are much less than one.

1. Severe Environmental Condition

This condition considers all the normal operating loads on the structure in combination with the loads resulting from an environmental event, such as wind or the OBE, which may occur only infrequently.

2. Abnormal Condition

This condition includes the Design Basis Accident Pressure Load (P_a) and the Design Basis Accident Thermal Load (T_a).

3. Extreme Environmental Condition

This condition includes loads resulting from environmental events which are credible but are highly improbable. These events include flood, the SSE, and the design tornado.

4. Abnormal/Severe Environmental Condition

This condition includes highly infrequent, simultaneous occurrence of abnormal and severe environmental effects.

5. Abnormal/Extreme Environmental Condition

This condition includes pipe rupture loads and direct pressure and jet impingement loads generated by a postulated rupture of high-energy piping. The condition is the highly improbable, simultaneous occurrence of abnormal and extreme environmental effects.

A summary of the nonservice load combinations is shown in Table 3.8.1-1.

3.8.1.3.3 Load Combinations on Localized Areas: Localized areas, such as penetrations, shell discontinuities, crane girder brackets, tendon and anchorage zones, and local areas of high thermal gradient, are designed for the same loading combinations as the Containment. In addition, local effects due to geometrical and mechanical discontinuities are considered.

3.8.1.3.4 Effect of Induced Strains on the Liner: Due to the prestressing forces and the DBA temperature effect in conjunction with other loadings, the steel liner plate is subjected to compressive stresses. In order to prevent instability and excessive deformation in the liner plate, continuous stiffeners are provided to anchor the liner to the concrete. The spacings of the stiffeners are determined such that the liner stresses and strains are in accordance with Section CC-3700 of the ASME-ACI 359 document.

3.8.1.3.5 Time-Dependent Effects: Time-dependent effects such as creep, shrinkage, steel relaxation, and other related effects, are considered in the design of the Containment.

3.8.1.3.6 Explanation of the Use of a Load Factor of 1.0: Nonservice load combinations that include extreme environmental effects, such as SSE or tornado effects, incorporate a load factor of 1.0 using a strength design approach with stresses within the range of general yield. This design approach is justified based on the fact that the extreme environmental effects that are considered are of an upper-bound conservative magnitude and have an extremely low probability of occurrence. The SSE is also assumed to occur concurrently with the DBA under the Abnormal/Extreme Environment Condition, an extremely unlikely occurrence. In addition, a margin of safety of at least 10 percent is provided in the DBA pressure.

3.8.1.3.7 Explanation for Load Factors:

1. Load Factors Under Service Load Category

The load factors of 1.0 used in the Service Load Category are conventional and are based on the working stress design method.

2. Load Factors Under the Nonservice Load Category

   a. The load factors under the Severe Environmental Condition: The load factors are in accordance with the ASME-ACI 359 document.

   b. The load factors under the Abnormal, Extreme Environmental, and Abnormal/Extreme Environmental Conditions.

      l) Dead Loads, Live Loads, Prestressing Loads, Operating Thermal Loads, and Operating Piping Loads (D, L, F, T_o, R_o) - These loads are accurately computable and are combined with an abnormal or extreme set of conditions which are not likely to occur. Therefore, a load factor of 1.0 is used.

2) DBA Pressure Loads (P_a) - For this load a factor of 1.5 is used for the first combination in the Abnormal Condition. A factor of 1.0 is used for the Abnormal/Extreme Environmental Condition. These factors are in accordance with current NRC positions.

3) Accident Thermal Pipe Loads (R_a) - Under the second combination of the Abnormal Condition, a factor of 1.25 is selected for R_a to assure sufficient margin of safety for intactness of pipe anchor embedments.

4) SSE Loads (E_{ss}) - The magnitude of acceleration chosen as representative of the most severe ground motion which could be postulated for this particular site. The intention of utilizing such a load is to demonstrate the functional capability of the structure; therefore, a load factor of 1.0 is chosen to meet this criteria.

5) Pipe Rupture Loads, Tornado Loads, Flood Loads, and DBA Thermal Loads, (Y, W_t, H, T_a) - A load factor of 1.0 is used with each of these loads because of their highly remote occurrence.

c. The load factors under Abnormal/Severe Environmental Condition are in accordance with current NRC positions.

3.8.1.4 Design and Analysis Procedures.

3.8.1.4.1 Analysis Procedures for the Containment: The Containment and its components are analyzed for all the load combinations described in Section 3.8.1.3.

3.8.1.4.1.1 Foundation Mat, Shell and Tendon Gallery Analyses - The Containment structure is analyzed with the BSAP computer program using a three dimensional finite element model that represents the shell, the hemispherical dome, the basemat and the effects of the internal structures. The containment is basically axisymmetric about its central vertical axis. Advantage is taken of building symmetry with only half the structure being modeled. Appropriate symmetric boundary conditions are imposed along the half model boundary line.

The foundation mat model incorporates the inclusion of the primary and secondary shield walls up to El. 19 ft, including the slab at the level. The effects of the remainder of the internals are represented by force boundaries at El. 19 ft-0 inches. Localized areas of discontinuity representing the sumps are considered in the model by appropriately reducing the stiffness characteristic of the elements. The coupling of the foundation media with the basemat is accomplished by using Winkler type soil springs. The magnitude of the spring constants vary for different loadings to account for the different characteristics of the loadings and their effect on the foundation media.

Both the basemat and the shell are modeled with plate elements. The complete model is shown in Figure 3.8.1-15. As can be seen from the model, the geometry at the shell basemat junction and at the apex of the dome require the utilization of a finer mesh. The discontinuities in the shell from penetration and buttress effects are neglected in the overall analysis of the shell since these do not affect the overall response of the structure. The discontinuities are analyzed in separate analysis.

Dead load is applied as a static gravity load. Prestressing load is established through a prestressing force analysis. The prestressing loads on the dome are computed by Bechtel standard computer program TENDON CE 239 (Appendix 3.8.A). These prestressing loads are input into BSAP model as nodal loads on the dome. The hoop tendon forces imposed on the containment wall are treated as axisymmetrical normal pressures on the wall. Design pressure load is applied as an outward pressure normal to the shell, dome and mat elements. Thermal loads (summer and winter) are obtained by subtracting the construction temperature (stress free temperature) from the average of surface temperatures given in Section 3.8.1.3.1. In addition, a linear gradient based on the difference of surface temperatures is considered. Accident temperature loading is considered as a non-linear profile in the analysis. No thermal gradient is considered for the basemat due to accident temperature loading because of the insulating effect of the two foot fill slab covering the mat liner. The effect of the hot liner on the concrete wall is considered in the design stage by using the OPTCON module of BSAP program.

Earthquake loads are applied as equivalent gravity accelerations on all structural elements for both horizontal directions and the vertical direction. Tornado loads are applied as normal pressures on the dome and cylindrical walls. The structural response for earthquake and tornado loads applied in the direction normal to the plane of symmetry are obtained by picking the response of an element at 90° azimuth angle to the same load applied in the direction of the plane of symmetry.

A summary of stress analysis results at key sections is shown in Table 3.8.1-7. Key sections are as indicated on Figure 3.8.1-14.

The tendon gallery is analyzed separately using manual methods. The top of the tendon gallery walls are considered fixed at the bottom of the containment, due to its relative stiffness. The design is performed using the loading combinations that are consistent with the loading combinations of the containment.

3.8.1.4.1.2 Equipment Hatch and Personnel Air Lock Analysis - The Containment shell is provided with a 24-foot-0-inch inside diameter opening for the equipment hatch and a 12-foot-1-inch inside diameter opening for the personnel lock. These openings give rise to stress concentration in their vicinity due to Containment loadings. The Containment wall is thickened in this region to accommodate higher stresses. For equipment hatch, the shell wall is thickened to 8 ft at the center line of the opening while for personnel air lock, wall thickness provided is 6 ft (Figure 3.8.1-4).

Stress analysis in the regions around equipment hatch and personnel air lock is based on finite element method using BSAP computer program and assuming that the concrete is elastic, isotropic, and homogeneous material. Post-tensioning tendons are draped around these penetrations. Effect of prestressing forces due to this tendon curvature in the plane of the shell wall is considered. For both openings, the finite element model includes at least an area within five times the radius of the penetration from the center of penetration, beyond which the effect of opening is assumed to vanish. The boundary conditions applied to the models are obtained from Containment shell analysis as described in Section 3.8.1.4.1.1. Figure 3.8.1-16 shows the boundaries of the mathematical model for equipment hatch opening analysis. Figure 3.8.1-17 shows the corresponding finite element mesh.

3.8.1.4.1.3 Buttresses and Tendon End Anchorage - Analysis and design of tendon and anchorage zones and reinforcement in buttresses are based on results of tests presented in Section 6.6 of BC-TOP-5A and conform to the requirements of the ASME-ACI 359, and Paragraph CC-3543. Refer to Figure 3.8.1-5 for buttress reinforcement.

3.8.1.4.1.4 Prestressing Force Analysis - The level of post-tensioning provided by prestressing tendons, after all predicted stress losses have taken place, is calculated by using a ratio of dead load plus prestress force to the accident pressure membrance force.

$$\text{ratio} = \frac{D + F}{Pa}$$

The ratio for vertical tendons: The critical section is at the apex of the dome; use = 1.3.

The ratio for hoop tendons: use = 1.2. The average effective force is calculated by using the lowest average stress obtained from one of the following:

- Average stress in any three adjacent tendons at the face of the buttress
- Average stress over the length of a tendon
- Average stress in any three adjacent tendons at a section consisting of the midpoint of any one tendon

The thickness of the dome and cylindrical wall is also checked to satisfy the allowable concrete compressive stresses. The initial membrane compressive stress of the net section before losses is limited to 0.35 f'_c, where f'_c is the specified compressive strength of concrete. The net section is considered to be the gross cross-sectional area less the area of tendon sheathing.

The post-tensioning forces acting on the Containment due to hoop tendons are treated as axisymmetric loads for the verification of the shell analysis as described in Section 3.8.1.4.1.1. The prestressing forces imposed on the dome by the two groups of vertical tendons and dome hoop tendons are calculated by the computer program TENDON, CE 239. (See Appendix 3.8.A for a detailed description).

3.8.1.4.2 Design Procedures for the Containment Structure: The design procedures and criteria for the Containment and its components, including the foundation mat and the steel liner plate, are in accordance with Article CC-3000 of the ASME-ACI 359 document with the exceptions described in Section 3.8.1.2.1. Computation of reinforcement is performed using the BSAP-POST program OPTCON module described in Appendix 3.8.A. Concrete is assumed cracked whenever tensile stresses are present. The cracked section analysis is performed for critical sections shown in Figure 3.8.1-14. Special design considerations are described below.

3.8.1.4.2.1 Steel Liner Plate and Anchorage System Design - The RCB is lined inside with a 3/8-in. welded carbon steel plate to ensure a vessel leaktight against the release of radioactive materials into the environment. The liner is also utilized as a concrete form during the construction stage. The liner plate has been thickened locally around penetrations and brackets up to a maximum 2-in. thickness.

The liner plate is anchored into concrete by a system of stiffeners welded onto the liner. In the cylinder region, the stiffeners are meridional angle and hoop channel sections, while in the dome region there are meridional tees and plates and hoop angles, as shown on Figure 3.8.1-6. A leak chase system is provided for inaccessible seam welds for monitoring the leak rate.

The computed stresses and strains in the liner consider the effect of the two-dimensional stress/strain field by use of the Poisson's ratio in stress and strain determination.

The spacing of meridional stiffeners is such that the compressive stress that would cause out-of-plane deformation of the liner exceeds the yield stress of the liner material. Due to fabrication tolerances, a condition of initial inward curvature may exist in some of the panels between stiffeners. Due to geometry change at such anchors, a condition of differential strain and hence a resultant shear will exist. All anchors are designed to resist this shear.

The force distribution between liner plate and anchors is based on a mathematical model consisting of a series of liner panels connected under applied compressive load. Each panel consists of anchors bearing against concrete and the liner plate in between anchors in tension or compression, each represented by elastic springs of respective stiffness. By this model any deformation in a panel is transformed into corresponding forces in the liner plate and anchors.

The spacing of anchors is such that the allowable stress and strain limits of Section CC-3700 of the ASME-ACI 359 document are not exceeded. Furthermore, anchors are designed such that if one anchor fails, the adjacent anchors are able to resist the additional loads to avoid a chain reaction failure.

The liner plate and anchors are analyzed for all the load combinations listed in Table 3.8.1-1, using load factors of 1.0. The ratio of energy available to energy used, called a factor of safety, is calculated for the anchors for all load combinations. Among all load combinations, a minimum factor of safety of 2.18 is obtained for cylindrical wall nonservice abnormal extreme environmental load condition.

3.8.1.4.2.2 Tendon Anchorage Zones - The design of tendon anchorage zones is in accordance with the requirements of ASME Section III Division 2, Paragraph CC-3543. The methodology of BC-TOP-5-A is applied.

3.8.1.4.2.3 Prestress Losses - In accordance with Section CC-3542 of the ASME-ACI 359 document, the design of the post tensioning tendons for the Containment considers the following effects:

- Final losses due to elastic shortening of concrete, concrete creep and shrinkage, and relaxation of tendon stresses are computed in accordance with the data contained in the paper, "A Method for Predicting Prestress Losses in a Prestressed Concrete Structure", by R. J. Glodowski and J. J. Lorenzetti.

- Friction losses due to intended or unintended curvature in the tendons are considered in accordance with the procedures described in Section CC-3542.2 of the ASME-ACI 359 document.

Except for the losses as specified above, further adjustments are considered to calculate the final effective prestress force at the end of plant life. They are:

- The provision of an additional 1.0 percent of steel area as an allowance for broken wires. Evidence of a broken wire during tensioning shall immediately be reported to the Engineer and made a part of the permanent stressing record. Loss due to breakage shall not exceed l percent in any three adjacent tendons.

- The ultimate tensile strength of a curved tendon is reduced by the resultant simultaneous application of lateral pressure. A 2-percent reduction in the ultimate tensile strength of the tendon is provided due to the assumed biaxial stress condition for all tendons.

All of the above losses are predicted with a reasonable degree of accuracy.

3.8.1.4.2.4 Design of Containment at Major Openings - Design of the Containment shell in the region of the equipment hatch and personnel air lock is based on the analysis results from Section 3.8.1.4.1.2. This region is in a state of biaxial stress.

In the thickened zone, circular reinforcement is provided for tangential, axial force, and moment. Grid reinforcement and/or radial reinforcement in the shell hoop and meridional directions is provided for radial axial force and moment.

Reinforcement for tangential shear force is provided in the principal axial directions according to the provisions of Section 3.8.1.4.2.5.1. Reinforcement for radial shear force is provided in the form of stirrups in accordance with Section 3.8.1.4.2.5.2.

Allowable stresses are given in Section 3.8.1.5.

3.8.1.4.2.5 Design for Shear Effects -

3.8.1.4.2.5.1 Tangential Shear -

1. Definition of Terms:

V_u and v_u = The peak membrane tangential shear force and stress, respectively, resulting from earthquake, wind, or tornado loading. When considering earthquake loading, the tangential shear force or stress shall be based on the square root of the sum of the squares of the multiple components of earthquake loading. For wind or tornado, the tangential shear force or stress shall be determined based on the direction of loading under consideration and shall be compatible with the determination of N_{he} and N_{ve}. The shear force shall be considered as positive and the units are k/ft and the shear stress has units of psi.

V_c = allowable tangential shear force carried by the concrete. The units are k/ft.

N_h and N_v = membrane force in the horizontal and vertical direction, respectively, due to pressure, prestress and dead load. N_h and N_v are positive when in tension and negative when in compression. The prestress force shall be the effective value.

N_{he} and N_{ve} = membrane force in the horizontal and vertical direction, respectively, from earthquake, wind, or tornado loading. When considering earthquake loading, the force shall be based on the square root of the sum of the squares of the multiple components of earthquake loading. When considering wind or tornado load, the force shall be based on the absolute sum of the horizontal and vertical components of loading. The force is always considered as positive.

N_{ht} and N_{vt} = membrane force in the horizontal and vertical direction, respectively, due to thermal effects.

The units of all preceding forces are in k/ft.

t = net wall thickness considering any reduction due to tendon duct in inches

b = unit length of section

f_y = yield strength of non-prestressed reinforcement, ksi

f'_c = compressive strength of concrete, psi ^{c}

2. Allowable Stresses

a. Nonservice Loads -

1) The applied tangential shear (V_u) shall not exceed 8.5bt $\sqrt{f'_c}$.

2) When both ($N_h + N_{ht} + N_{he}$) and ($N_v + N_{vt} + N_{ve}$) are compression, the allowable tangential force is:

$$V_c = [\,(N_h + N_{ht} + N_{he})\,(N_v + N_{vt} + N_{ve})\,]^{1/2}$$

(Eq. 3.8.1-1)

3) When V_u exceeds V_c, additional bonded reinforcing shall be provided in accordance with paragraph 3, below.

b. Service Loads

The applied tangential shear (V_u) shall not exceed 4.2bt $\sqrt{f'_c}$ and the expression for V_c shall be used as in paragraph a(2) above.

3. Design of Tangential Shear Reinforcing

a. Nonservice Loads

1) A sufficient amount of effective prestress shall be provided so the N_h and N_v are either compression or equal to zero.

2) When considering earthquake loading, the following equations shall be used:

$$A_{sh} = \frac{N_h + [N_{he}^2 + V_u^2]^{1/2}}{0.9f_y}$$

(Eq. 3.8.1-2)

$$A_{sv} = \frac{N_v + [N_{ve}^2 + V_u^2]^{1/2}}{0.9f_y}$$

(Eq. 3.8.1-3)

3) When considering wind or tornado loading in Eq 3.8.1-2 and 3.8.1-3, substitute N_{he} + V_u and N_{ve} + V_u for

$$(N_{he}^2 + V_u^2)^{1/2} \text{ and } (N_{ve}^2 + V_u^2)^{1/2}$$

where:

A_{sh} = area of bonded reinforcing steel in the horizontal direction (in.^{2}/ft)

A_{sv} = area of bonded reinforcing steel in the vertical direction (in.^{2}/ft)

b. Service Loads Design

The same requirements state under nonservice loads design shall be used in designing shear reinforcing for service load with the following modifications:

1) Equations 3.8.1-2 and 3.8.1-3 shall be replaced by:

$$A_{sh} = \frac{N_{he} + [N_{he}^{\ 2} + V^2]^{1/2}}{0.5f_y}$$

(Eq. 3.8.1-4)

$$A_{sv} = \frac{N_v + [N_{ve}^{\ 2} + V^2]^{1/2}}{0.5f_y}$$

(Eq. 3.8.1-5)

Where V is the applied tangential sheer, k/ft.

2) When considering wind or tornado loading in Equations 3.8.1-4 and 3.8.1-5, substitute N_{he} + V and N_{ve} + V for

$(N_{he}^{\ 2} + V^2)^{1/2}$ and $(N_{ve}^{\ 2} + V^2)^{1/2}$

3.8.1.4.2.5.2 <u>Radial Shear</u> -

1. Nonservice Load Design

a. The nominal shear stress, γ_u, shall be computed by:

$$\gamma_u = \frac{V_u}{0.85bd} \qquad \text{(Eq. 3.8.1-6)}$$

d need not be less than 0.85h for prestressed members.

b. When shear reinforcement perpendicular to the Containment surface is used, the required area of shear reinforcement shall not be less than

$$A_v = \frac{(\gamma_u - \gamma_c)\ bs}{f_y} \qquad \text{(Eq. 3.8.1-7)}$$

The perpendicular shear reinforcement shall not be spaced further apart than 0.50d.

where:

γ_c = Nominal permissible shear stress carried by concrete, psi.

c. When inclined stirrups or bent bars are used as shear reinforcement in reinforced concrete members, the following provisions apply:

1) When inclined stirrups are used, the required area shall not be less than

$$A_v = \frac{(\gamma_u - \gamma_c)bs}{f_y\ (\text{Sin}\ \alpha + \text{Cos}\ \alpha)} \qquad \text{(Eq. 3.8.1-8)}$$

2) When shear reinforcement consists of a single bar or a single group of parallel bars, all bent at the same distance from the support, the required area shall be not less than

$$A_v = \frac{(\gamma_u - \gamma_c)\ bd}{f_y(\text{Sin}\ \alpha)}$$

in which $(\gamma_u - \gamma_c)$ shall not exceed $3\sqrt{f'_c}$.

3) When shear reinforcement consists of a series of parallel bent-up bars or groups of parallel bent-up bars at different distances from the support, the required area shall be not less than that computed by Equation 3.8.1-8.

4) Only the center three-fourths of the inclined portion of any bar that is bent shall be considered effective for shear reinforcement.

5) Where more than one type of shear reinforcement is used to reinforce the same portion of the section, the required area shall be computed as the sum of the various types separately. In such computations, γ_c shall be included only once. The value of $(\gamma_u - \gamma_c)$ shall not exceed $8\sqrt{f'_c}$.

6) Inclined stirrups and bent bars shall be so spaced that every 45-degree line extending toward the reaction from the mid-depth of the section, 0.50d, to the tension bars shall be crossed by at least one line of shear reinforcement.

d. Shear reinforcement shall extend to at least a distance, d, from the extreme compression fiber and shall be anchored at both ends to develop the design yield strength of the reinforcement.

2. Service Load Design

The same requirements stated for the nonservice load design in this section shall be used in designing shear reinforcement for service loads with the following modifications:

a. Equation 3.8.1-6 shall be replaced by $\gamma = \dfrac{V}{bd}$

b. The reinforcement steel stress allowable from ASME-ACI 359 CC-3422.1 shall replace f_y in Equations 3.8.1-7, 3.8.1-8, and 3.8.1-9.

3.8.1.4.2.6 Methods of Providing Reinforcing Steel in Critical Areas - The methods of providing reinforcing steel in critical areas, such as in the buttresses and around the major penetrations and the smaller penetrations for pipelines, are depicted on Figures 3.8.1-4 and 3.8.1-5.

3.8.1.4.3 Evaluation of Effect of Variations in Assumptions and Materials: The fact that reinforced and/or prestressed concrete is not a homogeneous and isotropic material is accounted for in the design by the previously discussed considerations. Creep and shrinkage of concrete and other factors causing loss of prestress are considered in the design of the post-tensioning system by adjusting the required prestressing forces. The effect of an opening on the Containment shell is taken into account by utilizing a finite element technique to determine the increased forces and moments of the shell in the opening regions. Concrete cracking is considered in the design of reinforced concrete elements as discussed in Section 3.8.1.4.2. The stiffening effect of buttresses were considered per BC-TOP-5A.

3.8.1.5 Structural Acceptance Criteria. The Containment is designed to perform within the elastic range for the Service Load Category and is essentially elastic under the Nonservice Load Category. The allowable stresses and strains for the Service and Nonservice Categories are as follows:

3.8.1.5.1 Stresses for Service Loads – Working Stress Design:

3.8.1.5.1.1 Reinforcing Steel Allowable Stresses -

1. Bar Tension

   a. Average tensile stress: 0.5 f_y.

      The value given above may be increased by 33-1/3 percent when temperature effects are combined with other loads.

2. Axial Compression

   a. For load-resisting purposes, the allowable stress is 0.5 f_y.

      The value given above may be increased by 33-1/3 percent when temperature effects are combined with other loads.

   b. The stress may exceed that given in item 2.a for compatibility with the concrete but this stress may not be used for load resistance.

3.8.1.5.1.2 Concrete Allowable Stresses -

1. Concrete Normal Stresses

   a. Primary compressive stresses (as defined in Section CC-3136 of the ASME-ACI 359 document)

Membrane stress = 0.3 f_c'.

Membrane stress at initial prestress = 0.35 f_c'.

Membrane stress for load combinations including wind or earthquake = 0.40 f_c'.

Membrane plus bending = 0.45 f_c'.

b. Primary-plus-secondary compressive stresses (as defined in Section CC-3136 of the ASME-ACI 359 document)

Membrane stress = 0.45 f_c'.

Membrane plus bending = 0.6 f_c'.

c. Compression under the tendon anchor bearing plates is in accordance with Section CC-3421.1(d) of the ASME-ACI 359 document.

d. Concrete tensile strength is not relied upon to resist flexural and membrane tension.

2. Concrete Shear Stresses

a. Radial Shear Stresses

The allowable stresses and the limiting maximum stresses are in accordance with Section CC-3421.3 of the ASME-ACI 359 document.

b. Concrete Tangential Shear Stresses

Allowable stresses are given in Section 3.8.1.4.2.5.1.

3. Concrete Torsion and Bearing Stresses

The allowable stresses are in accordance with Section CC-3421.3 of the ASME-ACI 359 document.

3.8.1.5.2 Stresses for Nonservice Loads – Strength Design Method:

3.8.1.5.2.1 Reinforcing Steel Allowable Stresses and Strains -

1. Tension

   a. Average tensile stress is 0.9 f_y.

   b. The design yield strength of reinforcement is 60,000 psi.

   c. The tensile strain may exceed yield when the effects of thermal gradients through the concrete section are included.

2. Axial Compression

   a. For load-resisting purposes, the allowable stress is 0.9 f_y.

   b. The strains may exceed yield when acting in conjunction with the concrete if the concrete requires strains larger than the reinforcing yield to develop its capacity.

3.8.1.5.2.2 Concrete Allowable Stresses and Strains -

1. Concrete Normal Stresses

   a. Primary compressive stresses:

      Membrane stress = 0.6 f'_c.

      Membrane plus bending = 0.75 f'_c.

   b. Primary-plus-secondary compressive stresses:

      Membrane stress = 0.75 f'_c.

      Membrane plus bending = 0.85 f'_c with the limit of 0.002 in./in.

The stresses given above in items a and b are reduced, if necessary, to maintain the structural stability.

2. Concrete Shear Stresses

   a. Concrete Radial Shear Stress

   The allowable stress is in accordance with Section CC-3411.4.2 of the ASME-ACI 359 document.

b. Concrete Tangetial Shear Stress

The criteria for tangential shear are specified in Section 3.8.1.4.2.5.1.

3. Concrete Torsion and Bearing Stresses

The allowable stresses are in accordance with Section CC-3411 of the ASME-ACI 359 document.

3.8.1.5.3 Reinforcing Steel Requirements: The requirements for reinforcing steel splicing, anchorage, cover, and spacing are in accordance with Section CC-3530 of the ASME-ACI 359 document.

3.8.1.5.4 Concrete Crack Control: The requirements for crack control are in accordance with Section CC-3534 of the ASME-ACI 359 document.

3.8.1.5.5 Concrete Temperatures: Concrete temperatures do not exceed the values indicated in the ASME-ACI 359 document, Section CC-3430(a) for long-term loading and Section CC-3430(b) for accident or short-term loading.

3.8.1.5.6 Liners, Anchors, and Attachments – Allowable Stresses and Strains: The allowable stresses and strains in the liner plate are in accordance with the ASME-ACI 359 document, Table CC-3700-1. The allowable forces and displacements capacity of liner plate anchor are in accordance with Table CC-3700-2. The load categories shown in both tables include loads as defined in Section 3.8.1.3.

As stated in sections 3.8.1.4.1.4 and 3.8.1.3.4, containment prestress forces are sufficient to overcome DBA pressure and maintain the concrete containment and the liner in a state of compression, such that the concrete containment functions as an essentially leaktight barrier. The liner plate is a non-pressure retaining leaktight membrane wherever it is backed by concrete. Therefore, the fatigue analysis requirements of ASME-ACI 359, section CC-3760 are applicable only to the openings and penetrations designated as class MC components. CN-3054

3.8.1.5.7 Tendons Allowable Stresses: The tendon stresses at the anchor point do not exceed the allowable stresses described in Sections CC-3423 and CC-3413 of the ASME-ACI 359 document.

3.8.1.5.8 Design Criteria at the End of the Structure's Life: The design criteria at the end of the structure's life is the same as that described in previous sections. The prestressing load, F, considered in the load combinations includes the effective prestressing load at the end of the plant's life. It includes the effect of shrinkage of concrete, creep of concrete, relaxation of prestressing steel, elastic shortening of concrete, and seating of anchorage and friction loss due to curvature in the tendons. These effects can be reasonably predicted from past experience and research which has been done on prestress losses. These losses are verified by testing (see Section 3.8.1.4.2.3 for prestressing losses).

3.8.1.5.9 Effect of Repeated Reactor Shutdowns and Startups During the Plant's Life: Although the plant may be subjected to thermal cycling due to variation of temperature between shutdown and operating conditions of the reactor, it is unlikely that the margin of safety for concrete would be degraded. This is explained as follows. First, the stress due to thermal cycling is relatively small; secondly, the number of cycles of startups and shutdowns over the plant life is relatively small. Therefore, further consideration of fatigue effect in concrete is disregarded.

CN-3054

The effect of cycled stresses and strains in the liner is considered by performing a fatigue analysis, in accordance with Section 3.8.1.5.6, which includes the reactor shutdown-startup cycles.

3.8.1.6 Material, Quality Control, and Special Construction Techniques.

3.8.1.6.1 Concrete:

3.8.1.6.1.1 Materials –

1. Cement

The cement is in conformance with the requirements of ASTM C150-74, "Specification for Portland Cement", Type II, low alkali, moderate heat, and Section CC-2221 of the ASME-ACI 359 document. A typical summary of inprocess test results of cement appears in Table 3.8.1-2.

2. Aggregates

The aggregates are in conformance with the requirements of ASTM C33-74 and the following additional requirements.

a. Coarse aggregate gradations conform to:

   1) ASTM size no. 4 (1-1/2 in. to 3/4 in.)

   2) ASTM size no. 67 (3/4 in. to no. 4 mesh)

b. Limits on deletrious substances and physical properties of coarse aggregate comply with Table 3, "Moderate Weathering Region", tentative revision to ASTM C33-74.

c. Flat and elongated particles are limited to a maximum of 15 percent as defined and determined by CRD C119.

d. Abrasion loss when tested in accordance with ASTM C131-69 does not exceed 40 percent.

e. Fine aggregate gradation complies with ASTM C33-74.

f. Deleterious substances in the fine aggregate do not exceed the following:

1) 3.0 percent for clay lumps and friable particles

2) 3.0 percent for material finer than no. 200 mesh

3) 0.5 percent for coal and lignite

g. Fineness modulus of the fine aggregate is between 2.5 and 3.1.

The potential reactivity of the aggregates was evaluated in accordance with the Appendix to ASTM C33-74. The results of the evaluation indicate that the aggregates may be potentially reactive and therefore, in accordance with Paragraphs 4.3 and 8.2 of ASTM C33-74 and current industry practice, a low-alkali cement is being used.

The aggregates conform to the applicable requirements of Paragraph CC-2222 of the ASME-ACl 359 document as follows:

1) Subparagraph CC-2222.1, Sub-Subparagraphs a, d, e, and f

2) Subparagraph CC-2222.2, Sub-Subparagraphs a and b of the ASME-ACI 359 document

Aggregates for use in concrete are sampled and tested in accordance with Table CC-5200-1 of the ASME-ACI 359 document. Typical in-process test results of the aggregates appear in Tables 3.8.1-3A through 3.8.1-3E.

3. Mixing Water

The water used for mixing concrete and producing ice onsite complies with the requirements of Paragraph CC-2223, Subparagraphs CC-2223.1 and CC-2223.2, of the ASME-ACI 359 document and is supplied primarily from the deep aquifer through wells no. 5 and 6.

The chloride ion content of the water and ice used for mixing concrete does not exceed the limit of 250 ppm established in Subparagraph CC-2223.1 of the ASME-ACI 359 document.

When an additional ice source is utilized, the requirements of the referenced ASME-ACI 359 document are also complied with.

Water and/or ice for use in concrete is sampled and tested in accordance with Table CC-5200-1 of the ASME-ACI 359 document. Typical in-process test results appear in Table 3.8.1-4.

4. Admixtures

The admixtures used are in conformance with ASTM C260-73, "Standard Specification for Air-Entraining Admixtures for Concrete" and ASTM C494-71,

"Standard Specifications for Chemical Admixtures for Concrete (Type A and Type D)."

Pozzolans are not used in any concrete; however, a 90-day design strength requirement for Class A mixes has been selected to take advantage of later strength development and minimize the cement contents.

An air-entraining admixture is generally used in concrete for the primary purpose of enhancing workability. Durability considerations are minimal due to the geographic location of the plant site, thus permitting the air content of the individual mixes to be lower than normally recommended in order that strength is not adversely affected.

Water reducing (Type A) or water reducing and retarding (Type D) is generally used in concrete in order to minimize shrinkage, minimize the possibility of cold joints, permit reduced cement contents, and control the rate of heat rise.

In addition to the requirements of ASTM C260 and ASTM C494, the following requirements regarding the chloride ion content of the admixtures are applicable:

a. The chloride ion content of the admixture does not exceed 1 percent by weight of the admixture.

b. The chloride content of the admixtures is such that when the admixture is added to the concrete, the chloride content of the concrete is not increased by more than 5 ppm.

Typical in-process test results appear in Table 3.8.1-5.

3.8.1.6.1.2 Concrete Mixes: Selection of Concrete Mix Proportions - Proportions for concrete mixes are based on laboratory trial batches made of materials specifically approved for use and from which individual water/cement ratio curves were developed. Mix proportions are selected to ensure maximum workability and conformance with the concrete compressive strength requirements.

Proportions for the laboratory trail batches and the subsequent mix adjustments were in accordance with ACI 211.1-70, "Recommended Practice for Normal Weight Concrete."

Initially, concrete mix proportions were selected from the appropriate water/cement ratio curves, such that the average compressive strength exceeded f'_c; i.e., 5,500 psi (Class A) and 4,000 psi (Class B) by 1,200 psi. In addition, proportions were selected such that the plastic unit weight would not be less than 142 lb/ft^3 and the slump and air content would be 5 in. and 3 to 6 percent, respectively.

These initial mix proportions were used until sufficient test data (concrete cylinders tested in accordance with ASTM C39) became available and an over-design considerably less than 1,200 psi could be established.

The cylinder test data were analyzed in accordance with ACI 214, "Recommended Practice for the Evaluation of Compression Test Results of Field Concrete." A typical summary of in-process test data appears in Table 3.8.1-6.

New mix proportions were selected based on the water-to-cement ratio curves modified by field tests and the newly established over-design such that the requirements of Sub-Subparagraph CC-2232.2(b) of the ASME-ACI 359 document are complied with.

The durability of the concrete is not applicable as would be required for concrete subject to freezing and thawing. An air content less than required by Table CC-2232-1 of the ASME-ACI 359 document is used in order to obtain desired workability, and yet not reduce concrete strengths unnecessarily.

A maximum water-to-cement ratio of 0.48 is maintained for concrete placed below grade in order that permeability is minimized.

3.8.1.6.1.3 Concrete Properties - The concrete for the Containment shell has a minimum compressive strength of 5,500 psi at 90 days (Class A), and the concrete for the mat has a minimum compressive strength of 4,000 psi at 28 days (Class B).

The specified plastic properties are applicable at the point of placement. The targeted slump at placement is 3 in. with an allowable inadvertency margin of 2 inches. The air content range is 3 to 6 percent. Slump is determined in accordance with ASTM C143-71 and the air content is determined in accordance with ASTM C231-73.

Plastic unit weights are monitored in order that the required shielding characteristics of the concrete are achieved. Calculations for air dry unit weight of concrete were performed until a high degree of confidence was achieved that the in situ unit weight of the concrete is in excess of 136 lb/ft^3. A typical summary of in-process concrete test data appears in Table 3.8.1-6. The concrete and concrete constituents material properties compiled from subsequent, ongoing tests are maintained in a controlled project document.

Confirmatory tests to determine modulus of elasticity, Poisson's ratio, coefficient of thermal conductivity, coefficient of linear thermal expansion, length change (shrinkage coefficient), and density were performed on the mix proportions to provide actual property values for comparison with assumed design values. In addition, uniaxial creep, air dry unit weight, and apparent chloride content of the concrete were determined as modified by the concrete testing specification for a similar comparison.

3.8.1.6.1.4 Construction with Concrete - Concrete construction practices, including stockpiling, storing, batching, mixing, conveying, depositing, consolidating, curing, repairing, and the preparation of formwork and construction joints are in accordance with the provisions of Section CC-4200 of the ASME-ACI 359 document. The requirements of RG 1.55 are also complied with. No special construction techniques are utilized in the concrete construction.

3.8.1.6.1.5 Quality Assurance Programs - Quality assurance (QA) programs are established and implemented in accordance with ANSI N45.2. This meets the intent of Article CA-4000 of the ASME-ACI 359 document.

Specifically, QC programs are developed and implemented by the constructor, the concrete supply contractor and the testing contractor. These programs are monitored by the Construction Manager's QA organization.

1. The concrete supplier's program addresses and complies with Articles CC-2000, CC-4000, and CC-5000 as applicable.

2. The testing subcontractor's program addresses and complies with Articles CC-2000, CC-4000, and CC-5000 as applicable.

3. The constructor's program complies with Article CC-4000 and CC-5000 as applicable. The construction program also complies with the requirements of RG 1.55.

The QA program for the construction phase is described in the Quality Assurance Program Description. The QA program for the operations phase is described in the Operations Quality Assurance Plan.

3.8.1.6.2 Reinforcing Steel:

3.8.1.6.2.1 Materials -

1. Reinforcing Bars

All reinforcing bars are new billet steel conforming to the requirements of ASTM A615-72 Grade 60 and conform to the requirements of Sub-Subparagraphs CC-2310(a), CC-2331.2, CC-2332.2(a) and (b), and CC-2333(c) of the ASME-ACI 359 document.

2. Mechanical Splicing

Splice sleeves used for the mechanical splicing of reinforcing steel comply with the requirements of ASTM A519-73 Grades 1018 or 1026 or ASME SA-36. The splice sleeve material conforms to the requirements of Subparagraph CC-2310(b) and ASME SA-36 for sleeves which penetrate the liner and connect the dowels from the base mat to the primary and secondary shield wall. The liner plate is connected to the splice sleeves with a full penetration weld and reinforcing fillet welds (see B&R engineering report no. 2C829SR098-A).

3.8.1.6.2.2 Fabrication and Installation of Reinforcing Steel –

1. Fabrication of Reinforcing Steel

Fabrication of reinforcing steel complies with requirements of Subarticle CC-4300 of the ASME-ACI 359 document.

2. Installation of Reinforcing Steel

The splicing (mechanical and lap) of reinforcing steel is in compliance with Sub-Subarticle CC-4330 of the ASME-ACI 359 document. The installation of the reinforcing is in compliance with the Sub-Subarticle CC-4340 of the ASME-ACI 359 document.

All no. 14 and 18 reinforcing bars are spliced using standard Cadweld connectors and filler metal or the modified splice sleeves as described in Section 3.8.1.6.2.1.

3.8.1.6.2.3 Quality Assurance Programs - QA programs are established and implemented in accordance with ANSI N45.2. This meets the intent of Article CA-4000 of the ASME-ACI 359 document.

Specifically, QC programs are developed and implemented by the reinforcing steel supplier and the supplier of mechanical splice material. A portion of the construction QC program addresses reinforcing steel and mechanical splicing.

1. The reinforcing steel supplier's program addresses and complies with Articles CC-2000 (including special material testing, CC-2330) and CC-4000 of the ASME-ACI 359 document. The program addresses the requirements of RG 1.15 except as provided in the ASME-ACI 359 document.

2. The supplier of mechanical splice material has developed and is implementing a program which addresses and complies with the requirements of Articles CC-2000, CC-4000, and CC-5000 of the ASME-ACI 359 document.

The portion of the construction QC program applicable to the fabrication, splicing, placing, and testing of reinforcing steel and splice material addresses and complies with the requirements of Articles CC-2000, CC-4000, and CC-5000 of the ASME-ACI 359 document. The construction program addresses the requirements of RG 1.55 and complies with the requirements of RG 1.10, except as stated in Section 3.8.1.6.3.

The QA Program for the construction phase is described in the Quality Assurance Program Description. The QA program for the operations phase is described in the Operations Quality Assurance Plan.

3.8.1.6.3 Cadweld Splices:

1. All no. 14 and 18 reinforcing bars are spliced by the use of Cadweld connections, as described in Section CC-4333 of the ASME-ACI 359 document, to develop the tensile limits shown in Table CC-4330-1 of the document. The Cadweld splice design used has been presented to and reviewed by the NRC. The acceptability of this design is documented in NRC correspondence to HL&P dated March 24, 1977.

As an alternate to the requirements of RG 1.10, the provisions of the ASME-ACI 359 document, Paragraph CC-4333 are applicable as follows:

a. Subparagraph CC-4333.3, Initial Qualification Tests, serves as an alternate to Section C.1 of RG 1.10. In addition, a splicer will be requalified if in any 15 consecutive Cadweld tensile tests, two unacceptable Cadwelds are identified. The splicer will be requalified in the position or positions in which the failure(s) occurred. Qualification splices and procedures meet the requirements of Paragraph CC-4333 and Subsubarticle CC-5320. In addition, when an inspector finds that one individual performed two consecutive unacceptable Cadwelds in any one position, the responsible splicer shall be immediately located. The splicer will then perform the next two production splices for that position under 100 percent inspection (preparation and visual).

b. Sub-Subarticle CC-5320, Examination of Sleeves with Filler Metal Connections, serves as an alternate to Section C.2 of RG 1.10.

c. Sub-Subparagraphs CC-4333.4.2, Splice Samples, and CC-4333.4.4, Tensile Testing Requirements, serve as an alternate to Section C.3 of RG 1.10 except that the location of all Cadweld splices, including replacement splices, is maintained on "as built" sketches and additional records are maintained showing the location and test results of all splice samples tested. These records are in addition to the requirements of Subsubparagraph CC-4333.1.2, Maintenance and Certification of Records.

d. Sub-Subparagraph CC-4333.4.3, Testing Frequency, serves as an alternate to Section C.4 of RG 1.10, except that separate test cycles are established for each splicer as well as each position. The test frequency in CC-4333.4.3(a) is used throughout construction except when production splices are expressly prohibited by CC-4333.4.3(c), in which case straight sister splices are substituted on a one-for-one basis such that the 2 percent testing frequency is maintained.

e. Sub-Subparagraph CC-4333.4.5, Substandard Test Results, serves as an alternate to Section C.5 of RG 1.10 except that the designer rather than the constructor investigates the cause of failure in CC-4333.4.5(c).

2. Regarding the Cadweld splicing of no. 18 reinforcing bar dowels connecting the primary and secondary shield walls and other reinforced concrete internal structures with the RCB base mat, ASME SA-36 bar stock, aluminum-kilned, normalized, fine-grained material was selected due to its availability in bar stock of required size, its suitability for the purpose, and its favorable weldability qualities. The liner plate is attached to the Cadweld splice with a full penetration weld and reinforcing fillet welds, assuring a leaktight barrier.

3.8.1.6.4 Liner and Attachments:

3.8.1.6.4.1 Materials - Basic materials for the liner and attachments are as follows:

ASME OR ASTM SPECIFICATION DESIGNATIONS

Item	Materials
1. Liner Plate Materials	
a. Liner plate 5/8 in. thick or less	SA-285 Grade A
b. Liner plate thicker than 5/8 in.	SA-516 Grade 60
2. Anchorage and Stiffening Materials	
a. Stiffeners, embedded steel material, backing strips, and other miscellaneous metalwork	A-36 or A-516 or SA-285 Grade A
b. Stud materials	A-108 Grade 1010, 1015, 1016, 1018, or 1020
3. Spray Header Piping Anchors and Supporting Structure	SA-537 Class 1
4. Personnel and Auxiliary Airlocks and Equipment Hatch	SA-516 Grade 70 or SA-537 Class 1
a. Steel spacers, plates, and bars	A-36, A-366, SA-479, SA-516 Grade 70, SA-537 Class 1
b. Pipe couplings and plugs	A-105 or SA-105
c. Steel tubing	A-179
Item	Materials

d. Steel flanges, fittings, and pins	SA-182 SA-350 LF2
e. Hex bolts and tapping screws	A-193-B8 SA-193-B7 and B8 A-307
f. Hex nuts	A-194 Grade 8 SA-194 ZH
g. Steel bars and lock washers	A-276 Type 304 A-569, A-570 A-576
h. Material for class 2 air system	SA-213 Type 304
i. Stainless steel mating surface for seals	A-276 Type 304
5. Penetration Pipe Sleeves	
a. 6 in. to 24 inches in diameter	SA-333 Grade 6 or Grade 1 Seamless
b. Over 24 inches in diameter	SA-155 Grade KCF 60 Class 1 or SA-516 Grade 60 or 70
6. Emergency Sump Piping Sleeves	SA-106 Grade B Seamless or SA-333 Grade 6 Seamless
7. Bolts, Nuts, and Washers for Steam Generator and Reactor Coolant Pump Supports	
a. Bolts	SA-36
b. Nuts	SA-194 Grade 7
Item	Materials

c. Washers	SA-516 Grade 70
8. Polar Crane Girder, Bracket, Braces, Gussets, Stiffeners, and Bolts	
a. Bracket and girder	SA-537 Class 1
b. Braces, gussets and stiffeners	A-36
c. Bolts	A-490
d. Threaded rods	A-36
e. Nuts	A-194 Grade 2H
f. Washer	A-325
9. Cadweld Sleeves	
a. Attached with base liner	SA-36
b. All others	A615 or Approved Equal
10. Gaskets and Compressible Material	Ethylene Propylene Synthetic Rubber or Approved Equal
11. Penetration Gusset and Ring Plates	SA-516 Grade 60
12. Grounding Bars	SA-516 Grade 70
13. Welding Material	Per Section CC-2600 of ASME-ACI-359
14. Fuel Transfer Tube Sleeve (FTTS)	ASTM-SA-358 Grade 304, Class 1 SA-155 Grade KCF 60 Class 1 or, SA-516 Grade 60 or 70
15. FTTS Bellows	SA-240, Grade 304, Class 1

3.8.1.6.4.2 Special Material Testing and Examination -

1. Charpy Impact Testing

   Impact requirements for liner materials are as specified in NE-2320 and Section CC-2520 of ASME Code, Section III, as applicable. All specimens to be tested are Charpy V-notch, with the test temperature at least 30°F below the lowest service metal temperature, 50°F for liner material. Minimum impact values are as indicated in Table I-10.1, Appendix I, of ASME Code, Section III.

   The specimens for the anchor bolts meet the requirements of Table NF-2333-1. The test temperature for the anchor bolts is 50°F.

2. Lamination Testing

   Plates that are loaded during service in the through-thickness (short transverse) direction are examined in accordance with SA 578-73. The entire length and width of the plate is tested, using 9-in. gridlines.

   Also, special materials testing is required for the sections of the foundation mat liner plate located at the base of vertical members of the internal structure. The thickened plate is examined by ultrasonic testing to guard against any significant laminations. The seal welds of the Cadweld sleeves through the plate are inaccessible for leakage testing after the placement of concrete. Therefore, a test for leaktightness is performed in the shop before field installation. For this leakage test, a temporary channel is welded to the plate and the assembly is subjected to the same test pressure as that used in the Leak Chase Channel System for the foundation mat liner plate.

3.8.1.6.4.3 Fabrication, Installation and Welding of Liner - A fundamental requirement for fabrication and erection of liner plate is that welding procedures and welding operators are qualified by tests as specified in Section IX of the ASME B&PV Code, and as specified in Section CC-4500 of the ASME-ACI 359 document.

All temporary shoring and bracing furnished and installed by the vendor for erection of liner plates are subject to approval by the engineer.

The requirements of ANSI N45.2.2, Level D, for packaging, shipping, receiving, storage, and handling of items for nuclear power plants are complied with.

3.8.1.6.4.4 Examination of Liner - The nondestructive examination (NDE) of liner seam welds is in accordance with RG 1.19, "Nondestructive Examination of Primary Containment Liner Welds", and Section CC-5500 of the ASME-ACI 359 document.

3.8.1.6.4.5 Quality Control - Certified materials test reports are furnished by the steel liner vendor in accordance with the requirements of Section CC-2130 of the ASME-ACI 359 document.

Marking and identification of liner materials are in accordance with Section CC-2540 of the ASME-ACI 359 document.

Certification of tests and examinations are provided in accordance with the requirements of Section CC-4120 of the ASME-ACI 359 document.

The QA program for the construction phase is described in the Quality Assurance Program Description. The QA program for the operations phase is described in the Operations Quality Assurance Plan.

3.8.1.6.5 Post-Tensioning System:

3.8.1.6.5.1 Materials - The Containment uses a BBRV (Prescon Corp.) prestressing system. This system is one that has been reviewed and approved by the NRC in accordance with RG 1.103.

1. Tendons

The tendon is composed of 186 stress-relieved, high-strength wires of 1/4 in. diameter furnished in accordance with ASTM A421-77, type BA. The minimum ultimate strength of wire is 240,000 psi, with minimum yield strength of not less than 85 percent of the minimum ultimate strength.

The temporary corrosion prevention coating for tendons satisfies the requirements specified in Section CC-2442.2.2 of the ASME-ACI 359 document. The coating Visconorust 1601 Amber by Viscosity Oil Company is considered to be a qualified material.

The permanent corrosion prevention coating for tendons is a petroleum or microcrystalline wax-base material containing additives to enhance the corrosion-inhibiting and wetting properties, as well as to form a chemical bond with tendon steel. The properties of the coating and its chemical analysis limit are in accordance with Section CC-2442.3.2 of the ASME-ACI 359 document. The Visconorust 2090 P-4 by Viscosity Oil Company is considered to be a qualified material.

2. Buttonhead Anchorage

Buttonheads are cold-formed symmetrically about the axis of each wire and shall be free from harmful seams, fracture, or other flaws. The anchorage is assembled and shop buttonheaded. The anchorage assembly for the opposite end of each tendon is shop fabricated to permit rapid installation and button-heading in the field after the tendon has been placed in the structure.

The buttonheads and the anchorage assembly are fabricated with sufficient tolerance control that the anchorage assembly will develop the minimum breaking strength and required elongation of each individual wire and the tendon as a whole. The outside edge of any hole for a prestressing wire shall not be less than 1/4 in. from the root of a thread or from the edge of the assembly.

3. Bearing Plate and Trumpet Assembly

The bearing plate and trumpet are included as a part of the prestressing system and interact with the Containment at its interface.

The materials of individual components listed are given below:

Component	Material
Bearing Plate	ARMCO VNT plate per ASTM A633, Grade E
Trumpet	ASTM A283
Transition Cone	AISI 1008/1010
Extension Piece	AISI 1010/1018

4. Sheathing

Sheathings are used to provide a void in the concrete, wherein the tendons are placed, stressed, and greased. Duct materials are in accordance with Section CC-2441 of the ASME-ACI 359 document.

a. Semirigid Sheathing

The 5-1/4-in. outside diameter sheathing is a galvanized, spiralwrapped, semirigid, corrugated tubing which was made continuous by attaching a coupler to the ends of the sheathings. The coupler is a galvanized, semirigid, corrugated tubing approximately 1-ft-6-in. long. The internal diameter of the coupler is equal to the outside diameter of the sheathing. Leaktightness during concrete pour and greasing operations was maintained by wrapping each joint between the coupler and the sheathing with heavy-duty industrial tape. Drains were provided at all low points of the sheathing to prevent accumulation of water from condensation. The ends of the sheathing were kept closed by caps before the tendons were installed. Both semirigid sheathing and coupler material conform to ASTM A527.

b. Rigid Sheathing

The rigid sheathing is supplied with a rigid coupler to the trumpet capable of maintaining the required alignment. Rigid sheathing for the vertical, inverted, U-shaped tendon extends from the trumpet to a point 1 ft above the top of the base slab. For the hoop tendon, rigid sheathing extends from the trumpet to a tangent point.

5. Sheathing Filler Material

To prevent the migration of air and water to the tendon surface and sheathing void, the grease used for permanent protection is Visconorust 2090P-4, which meets the requirement of ASME Code, Section III, Division 2, Section CC-2442.3.2. This material is designed and certified by testing to be stable against physical and chemical changes for the life of the plant. Service temperature will change from 20° to 120°F. The expected integrated radiation doses are approximately 1.0 x 10^6R.

3.8.1.6.5.2 Fabrication and Installation - The fabrication and installation of the Post-Tensioning System are in accordance with Section CC-4400 of the ASME-ACI 359 document.

The tendon is handled, shipped, and stored in a manner that will not cause detrimental mechanical damage or corrosion to the material.

The fabrication of anchorage components is in accordance with Section CC-4431 of the ASME-ACI 359 document, which includes requirements for welding procedures and welder qualifications. The tendons are fabricated in continuous lengths without splices. The manufacturer establishes the methods and procedures for cutting tolerances, assembly procedures, and twisting and coiling of tendons.

A detailed installation procedure, including a checklist of work, is prepared before the tendon installation. The checklist includes lengths, locations, and numerical designations of the tendons, inspection and preparation of the tendon conduits, temporary corrosion protection of the tendons, requirements for welding or burning where tendons are handled and installed, and sequencing of installation. Tendon conduits are adequately supported against displacement during concreting. Their tolerances for position and alignment are specified. Open conduits are protected by capping or lugging to prevent entry of concrete or other foreign material. All joints are made tight against inleakage of mortar or appreciable water from the fresh concrete.

The tendon conduits are provided with a valve vent at the highest points of curvature to permit release of entrapped air pockets during greasing operation. Drains are provided at the lowest points of curvature to remove accumulated water prior to installing tendons. After the greasing process, the vents and drains will be closed and sealed.

The tendons are fabricated in continuous lengths without splices. All wires in a tendon are cut to the same length by cutting the wires under the same conditions. Welding for the anchorage components is performed using welding procedures and welders qualified in accordance with ASME B&PV Code, Section IX, 1974.

3.8.1.6.5.3 Tensioning Sequence - The detailed tensioning sequence is based on the design requirements to limit the membrane tension in the concrete to $1.0\sqrt{f'_c}$ and to minimize unbalanced loads and differential stresses in the structure. The post-tensioning procedure is prepared by the post-tensioning vendor, and the stressing sequence is established in that procedure.

The vendor is provided with effective prestressing force requirements for the Containment structure. Prestress losses due to concrete elastic shortening, shrinkage, creep, steel relaxation, and anchorage losses are considered. Forces and stress measurements are made by measuring the elongation of the prestressing steel and comparing it with the force indicated by the jack-dynamometer or pressure gage. Pressure gages or dynamometers are calibrated against known precise standards before their use in the prestressing operation, and all calibrations are so certified before use.

During stressing, records are kept of elongations as well as pressures obtained. Liftoff stress readings are taken at the end of each stressing operation to check the actual stress in the tendon. Dynamometer or gage readings are checked against elongation of the tendons, and any discrepancy exceeding ±5 percent of that predicted by calculations is resolved in consultation with the owner or his designated agent. The cause and resolution of the discrepancy is documented. Final elongation and stress are recorded.

3.8.1.6.5.4 Quality Control - The Post-Tensioning System vendor, Prescon Corp., established a record procedure which provides guidelines and requirements for the maintenance of QA records associated with the design, manufacture, tendon test and tendon placement. This document includes the following:

1. Quality control organization
2. Storage, preservation, and safekeeping
3. Control of fabrication of tendons and all other components (procurement and in-process control)
4. Installation inspection
5. Prestressing inspection
6. Final acceptance inspection
7. Control of nonconforming conditions
8. Reports, records and files

The QA program for the construction phase is described in the Quality Assurance Program Description. The QA program for the operations phase is described in the Operations Quality Assurance Plan.

3.8.1.7 Testing and Inservice Surveillance Requirements.

3.8.1.7.1 Structural Acceptance Test of Containment: Prior to initial fuel loading, the Containment is tested to a pressure equal to 1.15 times the Containment design pressure to ensure structural integrity under internal pressure. This test demonstrates that the Containment structure can resist the postulated accident pressure. In addition, by measuring structural response and comparing the results with analytical predictions, the test demonstrates that the structure behaves as predicted. RG 1.18, established a systematic approach to testing wherein quantitative information is obtained concerning structural response to pressurization.

The Containment structure is tested as a prototype containment in accordance with RG 1.18 criteria with the exception that the vertical, horizontal, and shear strains in the concrete are not measured under a prestressing anchor of a vertical tendon.

The Containment is subjected to an acceptance test that increased the Containment internal pressure from the atmospheric pressure to 1.15 times the Containment design pressure in approximately five equal pressure increments. The Containment is depressurized in the same number of increments. Strains and deflections are recorded at the atmospheric pressure and at each pressure level of pressurization and depressurization cycles. At each level, the pressure is held constant for at least l hour before the deflections and strains are recorded. Crack patterns are recorded at atmospheric pressure both before and immediately after the test and at the maximum pressure level achieved during the test.

In order to determine the overall deflection pattern of the Containment, the radial deflections of the Containment are measured at five points (exception to CC-6232 of ASME-ACI-359 document which requires a minimum of six points) along six meridians spaced around the Containment, including location, with varying stiffness characteristics, such as buttress, wall, and large opening. The vertical deflections of the Containment are measured at the apex, at six points along the spring line, and at two intermediate points between a point near the apex and the spring line on at least one meridian.

In order to determine the deflection pattern of the Containment wall adjacent to the largest opening, the radial and tangential deflections are measured at the equipment hatch at 12 equally spaced and symmetrically aligned points on the horizontal and vertical center axes.

The pattern of cracks that exceed 0.01 inches in width before, during, or after the test are mapped by 100 percent visual inspection near the base wall intersection, at mid-height of the wall, at the spring line of the dome, at one quadrant around the equipment hatch and personnel air lock, and at the intersection between the buttress and the wall. At each point, at least 40 ft^2 are mapped. The remainder of the Containment surface is inspected by high-power binoculars before pressurization to establish any initial cracks and after the test to determine any significant residual cracks. A visual inspection of the concrete in the area of accessible anchorage zones is made prior to the start of the test, at the maximum pressure level, and during the test.

The strain measurements in the concrete shell are made at the following location in the wall:

1. At the top of the base mat on two meridians, one meridian at a buttress and the second at the typical wall section away from discontinuities.

2. Around the equipment hatch at four locations symmetrically aligned on the horizontal and vertical axes (one location per quadrant, approximately 4 ft from the edge of the opening).

3. At mid-height of the cylinder on two meridians, one at a buttress and the second at the typical wall section.

4. At the level of the spring line on two meridians, one at a buttress and the second at the typical wall section.

For each of the above locations, the strain measurements are made at three positions within the wall (i.e., near the inside face, approximately at mid-point, and near the outside face). Horizontal, vertical, and diagonal strains in the concrete are measured at each position.

The location of the deflection points, strain gages, and crack inspection areas are shown on Figure 3.8.1-13.

Temperature and strain measuring devices shall be recorded 24 hours prior to the starting of pressurization at 3-hour intervals (exception to CC-6235 and CC-6242 of ASME-ACI-359 document which require reading being taken one week prior to starting).

3.8.1.7.2 Integrated Leak Rate Test: The integrated leak rate test is as described in Section 6.2.6.

3.8.1.7.3 Inservice Surveillance Program:

3.8.1.7.3.1 Inservice Surveillance of Ungrouted Tendons - The inservice surveillance of ungrouted tendons complied with the requirements of RG 1.35 through the tenth year surveillances. The fifteenth and twentieth year surveillances complied with the 1992 Edition 1992 Addenda of ASME Section XI, Subsection IWL, as modified and supplemented by 10CFR 50.55a(b)(2)(viii). The twenty-fifth and thirtieth year surveillances will comply with the 2004 Edition No Addenda of ASME Section XI, Subsection IWL, as modified and supplemented by 10CFR50.55a(b)(2)(viii).

3.8.1.7.3.1.1 Tendon Prestress-Level Surveillance - Tendon liftoff tests to monitor loss of prestress are performed using properly calibrated jacks. Provisions are made to ensure that the elongation and the jacking force are measured simultaneously. Tolerances for possible discrepancies between the elongation and the jacking force are established in the design specifications. The maximum probable error in the liftoff test results and the accuracy achieved during the test are evaluated. The probable influence of temperature on the test results due to change in the length of the wires, size of the structure, and changes in friction values is evaluated.

The procedure for the liftoff test is in accordance with the following:

1. A measurement of the prestress at liftoff.

2. Increase of the liftoff force up to a value greater than the expected maximum value of prestressing force.

3. Unloading of the tendon to complete detensioning (zero tension).

4. Examination for evidence of steel failure. (Broken wires shall be removed.)

Items 1 and 4 apply to all surveillance tendons, while items 2 and 3 apply to one tendon from each group (inverted U and hoop).

The acceptance criterion for individual tendon prestress loss is that the tendon has a prestress force not less than the predicted lower bound of prestress force for the time of the test.

3.8.1.7.3.1.2 Tendon Material Surveillance - The following numbers and types of previously stressed tendon wires are removed from the following tendon groups for test and examination to detect evidence of corrosion or other deleterious effects:

1. U-shaped tendon - one; at each successive inspection, a sample is selected from a different family of tendons

2. Hoop tendon - one

The tensile tests are made on at least three samples cut from each removed wire (one at each end and one at mid-length). The length of the samples is practical for testing. The use of fatigue tests and accelerated corrosion tests are considered where applicable.

3.8.1.7.3.1.3 Anchor Surveillance - Hardware, such as bearing plates, stressing washers, shims, and buttonheads, are visually inspected to the extent possible without dismantling load-bearing components of the anchorage.

3.8.1.7.3.1.4 Sheathing Filler Surveillance - The method to be used for checking the presence of sheathing filler grease accounts for the following:

1. The minimum coverage permitted for different parts of the anchorage system, including buttonheads.

2. The influence of temperature variations, especially the lowest temperature likely to occur between the successive inspections.

3. The procedure used to uncover possible voids in grease in the trumpet.

4. Grease specifications, qualification tests and acceptability tolerances.

The removal of grease to permit visual inspection of the stressing washers, shims, and bearing plates does not increase the effects of corrosion or damage the steel.

3.8.1.7.4 Containment Inservice Inspection Requirements

3.8.1.7.4.1 Components Subject to Examination and/or Test – ASME Code Class MC and metallic liners of Class CC components will be examined and tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code as required by 10 CFR 50.55a, except where specific written relief has been requested. Additional requirements related to the inspection of Class MC and metallic liners of Class CC components are imposed by 10 CFR 50.55a (b)(2)(ix).

3.8.1.7.4.2 Accessibility – Accessibility to containment features was not required by 10 CFR 50.55a at the time of construction. Containment accessibility will be maintained to the extent practical during subsequent modifications.

3.8.1.7.4.3 Examination Techniques and Procedures – Examination techniques and procedures for Class MC and metallic liners of Class CC components will be in accordance with Articles IWA-2200 and IWE-2000 of the ASME Section XI.

3.8.1.7.4.4 Inspection Intervals – The inspection interval for Class MC and metallic liners of Class CC components will be in accordance with IWE-2000 of ASME Section XI.

3.8.1.7.4.5 Examination Categories and Requirements – The examination categories and requirements for Class MC and metallic liners of Class CC components will be in agreement with IWE-2000, respectively, of ASME Section XI and the additional requirements imposed by 10 CFR 50.55a.

3.8.1.7.4.6 Evaluation of Examination Results – Evaluation of examination results for Class MC and metallic liners of Class CC components will be performed in accordance with Articles IWE-3000, respectively, of ASME Section XI. Additional evaluation and reporting requirements are imposed by 10 CFR 50.55a.

3.8.1.7.4.7 System Pressure Tests – Class MC and metallic liners of Class CC components subject to system pressure tests will be tested in accordance with IWE-5000, respectively, of ASME Section XI.

3.8.2 Steel Containment System (ASME Class MC Components)

This section, as outlined in the NRC format regarding a "Steel Containment", is not applicable to the STPEGS Containment structure itself, since a steel-lined, post-tensioned concrete Containment is used, as described in Section 3.8.1.1. However, certain steel items in the Containment System are designed, fabricated, and installed in accordance with the intent of the technical requirements of the ASME B&PV Code, Section III, Class MC Components. These items, as described in Section 3.8.1.1, consist of the following:

1. Personnel and auxiliary airlocks

2. Equipment hatch

3. Other penetrations subject to pressure-induced stresses

This section addresses itself to the requirements of the ASME Code, Section III, Class MC Components. The personnel and auxiliary airlocks are tested and receive a nameplate with an N symbol. The equipment hatch and other penetrations are not stamped because they are an integral part of an unstamped Containment vessel. The equipment hatch and air lock attachment collar welds will be tested during the Structural Integrity Test of the Containment Structure.

3.8.2.1 Description of ASME Class MC Components. Access into the RCB is provided by an equipment hatch, personnel and auxiliary airlocks, and penetrations.

3.8.2.1.1 Equipment Access Hatch: The equipment hatch consists of a removable flanged head, matching body ring, swing bolts, and seals. The body ring has a 24-ft inside diameter and is stiffened on its exterior surface by a welding collar, designed for attachment by welding to a thickened insert plate in the RCB liner. The body ring is anchored into the concrete Containment wall.

The swing bolts are provided and installed on the body ring. These are to be used with matching brackets on the head to draw the head tight and to provide an effective seal. Two concentric grooves are machined in the flanged head to accept two separate 0-ring seals.

The head is flanged to match the body ring and is of a dished shape that is convex to the pressure. Brackets for accepting the body ring swing bolts are provided on the outside diameter of the flange. A test connection is provided between the two concentric seal grooves in the head for shop leak-testing between the two 0-ring seals and for future field testing.

The dished head is fully removable by a vertical lifting device. The head runs in guides throughout the extent of its vertical movement. The guides are securely fixed through the liner plates at sufficient positions to ensure the rigidity of the assembly. A locking device on each guide is provided to support the head in its raised position. For typical details of the equipment hatch, refer to Figure 3.8.2-1.

3.8.2.1.2 Personnel and Auxiliary Airlocks: The personnel airlock is a double, inflatable seal airlock, and the auxiliary airlock is a double compression seal airlock. The personnel airlock is provided with an air supply system, a pressure equalizing system, a leak rate monitoring system, and an electrical and instrumentation systems. The auxiliary airlock has a pressure equalizing system and electrical and instrumentation systems. The personnel airlock air supply system has two Class 2 air tanks per door and provides complete redundancy required to meet single failure criteria. Each seal has its own airtank and check valve to supply air to the seal in case of loss of plant air. A separate hydraulic system is provided to operate each door. The airlock barrels are inserted through existing Containment wall sleeves; then the attachment collars furnished with the airlocks are welded to the sleeves. The personnel airlock barrel has an 11-foot-6-inch inside diameter with sufficient length to provide a minimum clear distance of 8 ft between doors.

The personnel airlock has two gasketed doors in series. The clear opening of the door is 5 ft wide by 8 ft high. The personnel airlock door seals can be leak tested by pressurizing the area between the seals through a pipe tap which is located in each door. The entire airlock can be leak tested by pressurizing through the emergency air supply. The personnel airlock is designed so that if a DBA occurs, the pressure will seal the doors into the airlock frame. When the entire airlock is leak tested, the pressure is forcing the inner door (reactor end) open, or into the unseated position. In order to prevent the airlock door from being unseated, test clamps and lugs (strong backs) are provided to hold the reactor end door in place during leak testing. A list of the mechanical and electrical penetrations of the personnel airlock is provided in Table 3.8.2-2.

The auxiliary airlock barrel is 10-ft long with a 5-ft-6-in. outside diameter. A 30-inch-diameter door is located at each end of the auxiliary airlock. Each door is hinged and furnished with seals mounted on the door and impinged upon stainless steel surfaces. The space between the double seals on each door is capable of being pressurized to the design pressure without the use of the test clamps. Both seals must be leaktight under this condition. The pressurization of this space will create a pressure barrier at each door which is automatically sequenced into the normal door operation.

The two doors for the personnel airlock are electrically and mechanically interlocked so that one door cannot be opened unless the second door is sealed. The doors for the auxiliary airlock are mechanically interlocked. Provisions are made to bypass the interlock to permit both doors to be opened when safe to do so.

A pressure-equalizing valve at each door is provided to equalize pressure across the doors when personnel are entering or leaving the Containment. The valves are properly interlocked so that they both cannot open at the same time, and each valve can be operated only when the opposite door is closed and locked.

The air supply solenoid valves (located outside Containment) are closed upon receipt of a Containment isolation Phase A signal. These lines are considered Containment penetrations and are detailed in Figures 6.2.4-1. They are designed in accordance with 10CFR50, Appendix A, GDC 57 and tested in accordance with 10CFR50, Appendix J.

3.8.2.1.3 Penetrations: Other penetrations through the Containment include the electrical penetrations, piping penetrations, and the fuel transfer tube sleeve. All penetrations are pressure-resistant, leaktight, and welded assemblies. The penetrations are welded to the liner and anchored into the concrete wall of the Containment.

3.8.2.1.3.1 Electrical Penetrations - Typical electrical penetration is shown on Figure 3.8.1-12. Design details are discussed in Section 8.3.

3.8.2.1.3.2 Piping Penetrations - Single-barrier piping penetrations are provided for all piping passing through the Containment wall. The closure of the pipe to the steel liner is accomplished with flued heads, pipe caps, or plates buttwelded to the pipe and penetration sleeve. In the case of piping carrying hot fluid, the pipe is insulated. Figure 3.8.1-11 shows a typical high-energy line penetration. For single pipe penetration for moderate-energy lines, see Figure 3.8.1-10. The MC classification extends from the containment liner to the flued head or cap of the penetration.

3.8.2.1.3.3 Fuel Transfer Tube - A fuel transfer penetration is provided for fuel movement between the refueling canal in the Containment and the fuel transfer canal in the FHB. The penetration consists of a 20-in. outside diameter stainless steel pipe that acts as the transfer tube, and is fitted with a double-gasketed blind flange in the refueling canal and a standard gate valve in the fuel transfer canal. The casing stainless steel pipe is provided with expansion bellows and is connected to the Containment steel liner penetration. The transfer tube sleeve assembly is fitted with a test connection which permits local leakage testing of the expansion bellows. For typical details, see Figure 3.8.1-8.

3.8.2.1.4 Design Bases: Containment penetrations are designed to maintain Containment integrity during normal operation of the plant and in the event of a DBA. All Containment penetrations are designed to meet the intent of the Class MC components of the ASME B&PV Code, Section III. Penetrations are designed in accordance with NRC General Design Criterion (GDC) 53 of 10CFR50, Appendix A and, in addition, are designed to meet the following considerations:

1. Ability to withstand the maximum design pressure that can occur due to the postulated rupture of any pipe inside the Containment.

2. Ability to withstand the jet forces associated with the flow from a postulated rupture of the pipe in the penetration and maintain the integrity of the Containment.

3. Ability to accommodate thermal and mechanical stresses encountered in normal operation and other modes of operation and testing.

The anchorages of all penetrations to the Containment wall are designed as Category I structures to resist all forces and moments caused by a postulated pipe rupture, and thermal and seismic loads. The penetration assembly welds and welds to the liner are full penetration welds.

3.8.2.2 Applicable Codes, Standards and Specifications.

3.8.2.2.1 Basic Code: The basic code for the design, materials, fabrication, testing, and examination of these steel items is the ASME B&PV Code for Nuclear Power Plant Components, Section III, Subsection NE, for Class MC Components.

3.8.2.2.2 Other Applicable Codes, Standards and Specifications: These additional codes, standards and specifications and Government regulations as discussed in Section 3.1 are applicable to the construction, inspection, materials, and testing of the Class MC steel components:

1. ASME Code, Sections II, III Division l, and IX, 1971, including winter 1973 addenda (excluding fuel transfer tube sleeve, personnel and auxiliary airlocks)

2. ASME Code, Sections II, III Division l, and IX, 1974, including winter 1975 addenda (for fuel transfer tube sleeve and personnel and auxiliary airlocks)

3. Standard Specifications for Electric-Fusion Welded Austenitic Chromium-Nickel Steel Pipe for High-Temperature Service - ASTM A358-1975

4. ASME Code, SA-240, Grade 304, Stainless Steel Material

5. ANSI N45.2, Quality Assurance Program Requirements for Nuclear Power Plants, 1971

6. NRC RG 1.57, Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components

3.8.2.2.3 Exceptions: The exceptions to the ASME B&PV Code, Section III, are:

1. Field installation - The requirement for a Certificate of Authorization per Article NA-8231 is excluded. The field work covered by this exception is limited to the welding of the personnel and auxiliary airlock collars, and electrical assemblies to the embedments in the shell, and installation of the equipment hatch.

2. Shop fabrication of equipment hatch and other penetration sleeves - The requirements for Authorized Inspection Agency per Article NA-5000, and Nameplates, Stamping and Data Reports per Article NA-8000 are excluded.

3. Fuel transfer tube sleeve (FTTS) - The testing to verify the leaktight integrity of the FTTS after installation is allowed to be either by a hydrostatic test in accordance with the requirements of Article NE-6220 or by a pressure decay test. Testing exclusively by a hydrotest in accordance with the Code is not a mandatory requirement since the FTTS has been specified as not requiring the Code N-stamping for Class MC items.

All of the foregoing components will be subject to verification by pressure testing during the Structural Integrity Test and the Integrated Leak Rate Test in accordance with 10CFR50, Appendix J.

3.8.2.3 Loads and Load Combinations. ASME Code, Section III, Division 1, Subsection NE and RG 1.57 are not explicit with respect to the loads and load combinations which should be considered in the design of Class MC items. All applicable loads as listed and defined in Section 3.8.1.3 are considered:

D --- Dead loads

L --- Live loads

F --- Prestress loads

P_t -- Test pressure

T_t -- Test temperature

T_o -- Thermal effects and loads during startup, normal operating and shutdown conditions

R_o -- Piping reactions during startup, normal operating and shutdown conditions

E_o -- Loads generated by the OBE

E_{ss} - Loads generated by the SSE

P_a -- Design Basis Accident Pressure Load

T_a -- Design Basis Accident Thermal Load

R_a -- Pipe accident reaction

Y -- Equivalent static load on the component generated by the reactions on the broken pipe, jet impingement and missile impact during the DBA.

P_v -- Subatmospheric pressure load (external pressure)

The load combination utilized in the design of Class MC items is shown in Table 3.8.2-1.

3.8.2.4 Design and Analysis Procedures. The Class MC items are analyzed and designed in accordance with the applicable requirements of ASME Code, Section III, Subsection NE. The analysis and design of the equipment hatch, personnel airlock, and auxiliary airlock are performed by a selected vendor using appropriate conventional engineering methods.

3.8.2.5 Structural Acceptance Criteria. The structural acceptance criteria for Class MC items are in accordance with Article NE-3000 of Section III of the ASME Code. The design is such that all the stress and strain limits defined in Article NE-3000 are satisfied for pressure loads in combination with all mechanical loads and thermal loads.

The requirements of RG 1.57 are complied with.

3.8.2.5.1 General Criteria: The ASME Code, Section III, Subsection NE design criteria for Class MC items are based on establishing stress and strain limits which vary according to the following factors:

1. Type of stress, such as primary stress, secondary stress, and peak stress.

2. Type of stress component, such as membrane stress and bending stress.

3. Type of load, such as mechanical and thermal loads.

(For the definition of these stresses and loads, refer to Article NE-3000 of Section III of the ASME Code.)

3.8.2.5.2 Allowable Primary Stress Intensities:

3.8.2.5.2.1 Considering Mechanical Loads Only - Based on the load combinations of Section 3.8.2.5.4 (1), the following allowable stress intensities, in accordance with Section NE-3131(a), (b), and (d), and Section NE-3133 of Section III of the ASME Code, are complied with:

1. General Primary-Membrane Stress Intensity -
   Allowable value = 1.0 x S_m

2. Local Primary-Membrane Stress Intensity -
   Allowable value = 1.5 x S_m

3. Primary-Membrane-Plus-Primary-Bending Stress Intensity -
   Allowable value = 1.5 x S_m

The exception provided by Section NE-3131(d), "In considering the provisions of NE-3222.4(d) consideration need not be given to the effects of earthquake loading", should not be applied to load combination A.

(The design stress intensity values, S_m, are in accordance with Section NE-3229 of Section III of the ASME Code.)

3.8.2.5.2.2 Considering Safe Shutdown Earthquake - For load combinations B and E of Table 3.8.2-1, which include the effects of the SSE, regions where the structure is integral and continuous may have higher allowable stresses in accordance with Section NE-3131(c)(2).

3.8.2.5.2.3 Considering Effects of a Pipe Rupture Load - Load combination F, which includes the effects of a pipe rupture load, Y, is evaluated in accordance with Section NE-3131.2 of Section III of the ASME Code.

3.8.2.5.3 Allowable Primary-Plus-Secondary Stress Intensities:

3.8.2.5.3.1 Considering Mechanical-Plus-Thermal Loads - Based on the load combinations included in Section 3.8.2.5.4 (2), the allowable stress intensity value = 3.0 x S_m, in accordance with Sections NE-3131(b) and NE-3222.2 of Section III of the ASME Code.

3.8.2.5.3.2 Test Requirements - The design limits of Section NE-6000 are applied for load combination G in Table 3.8.2-1 for the tests stipulated by Section NE-6000. For tests in addition to the l0 tests permitted by Section NE-6000, the design limits of Section NE-3226 (a), (b), and (c) and Section NE-3131(d) are applicable.

3.8.2.5.4 Design Loading Combinations:

3.8.2.5.4.1 Primary Stresses - For the loading combinations to be considered in evaluating the primary stresses, refer to load combinations B, C, D, E and F in Table 3.8.2-1.

3.8.2.5.4.2 Primary-Plus-Secondary Stresses - For the loading combinations to be considered in evaluating primary and primary-plus-secondary stresses, refer to load combinations A and G in Table 3.8.2-1.

3.8.2.5.5 Miscellaneous Considerations:

3.8.2.5.5.1 Compressive Stresses - In areas of compressive stresses, buckling criteria are considered in accordance with Article NE-3000 of Section III of the ASME Code.

3.8.2.5.5.2 Plastic Analysis - Strains associated with primary-plus- secondary stress intensities may be exceeded if a plastic analysis is performed and if the requirements of Section NE-3228 of Section III of the ASME Code are complied with. This approach may be required when considering the differential thermal growth, due to an accident temperature, of a penetration sleeve that is partially encased in the Containment wall. In such a situation, since only one (or several, at most) cycle of accident temperature need be considered, shakedown (as defined in Section NE-3213.18 of Section III of the ASME Code) is not a consideration.

3.8.2.5.5.3 Fatigue Analysis - The requirements for an analysis of cyclic operation are investigated in accordance with Sections NE-3222.4 and NE-3131(d) (and the referenced sections therein) of Section III of the ASME Code.

3.8.2.6 Materials, Quality Control and Special Construction Techniques.

3.8.2.6.1 Materials: The materials utilized for Class MC items are in accordance with Article NE-2000 of Subsection NE of the ASME Code, Section III, Division 1. The following materials are used.

3.8.2.6.1.1 Carbon Steel Plates - Carbon steel plates conform to ASME SA-516, Grade 70 and SA-537, Class 1.

3.8.2.6.1.2 Penetration Pipe Sleeves -

1. A diameter of 6 in. to 24 in. conforms to ASME SA-333, Grade l and SA- 333, Grade 6.

2. Over 24 inches in diameter conforms to ASME SA-155, Grade KCF 60, Class 1. Rolled and welded pipes conform to SA-516, Grade 60 and Grade 70.

3. Emergency sump piping sleeves conform to SA-106, Grade B or SA-333, Grade 6.

4. Gasket materials are Ethylene Propylene Synthetic Rubber or approved equal.

5. Penetration gussets and ring plates conform to SA-516, Grade 60.

6. Stiffeners conform to SA-516, Grade 70.

7. Equipment hatch-swing bolts conform to SA-193, Grade B7.

8. Stainless steel bellows to conform to SA-240, Grade 304.

For protective coatings, refer to Section 6.1.2.1.

3.8.2.6.2 Quality Control: The QA program for the construction phase is described in the Quality Assurance Program Description. The QA program for the operations phase is described in the Operations Quality Assurance Plan. In addition, the particular QC measures which are required for the ASME Code Class MC items are outlined below.

1. The vendor supplying Class MC items submits shop and field quality compliance or QA organization and procedures. These procedures include, as applicable, the methods of documentation of materials, material control, welder identification, and welding electrode handling and distribution. Further, the vendor submits methods of qualification of NDT and welding personnel, procedures, and equipment.

2. The records pertaining to the Class MC items contain three distinct categories: materials certifications, welding data, and test data. All records are turned over to the owner on completion of the work.

3. All welding procedure qualifications and welder performance qualifications are in accordance with ASME Code Section IX. The welding design, fabrication, inspection, and acceptance conform, as a minimum, to the requirements of ASME Code Section III, Subsection NE. The examination of welds for Class MC items is in accordance with Article NE-5000 of Subsection NE of the ASME Code, Section III, Division 1.

4. All procedural requirements for nondestructive testing (NDT) conform, as a minimum, to the requirements of Appendix IX of Section III of the ASME Code.

3.8.2.6.3 Special Construction Techniques: No construction techniques unusual to current methods are used for the Class MC items.

3.8.2.7 Testing and Inservice Surveillance Requirements. The personnel and auxiliary airlocks are shop tested upon completion in accordance with ASME Code Section III, Subsection NE, Article NE-6000 requirements, and each has a nameplate with the N symbol for Class MC components.

The Class MC items are subjected to the structural acceptance test as described for the Containment in Section 3.8.1.7.1.

Following the successful completion of the structural acceptance test, the leak-rate test described in Section 6.2.6 is performed with the personnel airlock and equipment hatch inner doors closed. The design pressure is maintained for whatever length of time is required to demonstrate full compliance with the leaktightness requirements.

In addition, upon completion of construction, the personnel airlocks and the equipment hatch are given an operational test consisting of repeated operation smoothly without binding or other defects. All defects encountered are corrected and retested. The process of testing, correcting defects, and retesting is continued until no defects are detectable.

Preoperational and periodic leak tests of the testable penetrations are conducted to verify their continued leaktight integrity below the specified design leak rate. These tests are discussed in

Sections 6.2.6. Certain valves in the airlocks are within the scope of the ASME Inservice Testing Program and are tested in accordance with requirements of Section 3.9.6.2.

3.8.3 Concrete and Structural Steel Internal Structures of Concrete Containment

3.8.3.1 Description of the Containment Internal Structures. The Containment internal structures are designed to provide structural supporting elements for the major components of the Nuclear Steam Supply System (NSSS) as well as to provide required shielding, both against internal missiles and for biological protection. Basic structural components are designed using both reinforced concrete and structural steel as appropriate. The internal structures consist of the following major elements. For details of the internal structure arrangement, refer to the general arrangement drawings listed as Figures 1.2-12 through 1.2-20 in Table 1.2-1.

3.8.3.1.1 Primary Shield Wall: The primary shield is a 7-ft nominal thick, heavily reinforced, concrete wall, shaped as an octagonal-prism, with a cylindrical core removed to house the reactor pressure vessel (RPV). The primary shield wall is situated at the center of the RCB 1 ft off the east-west centerline, and extends up from the interior base slab at El. (-)11 ft-3 in. to the refueling pool at El. 38 ft-6-1/2-inch. It is built integrally with the refueling cavity walls extending up to El. 68 ft-0 inch. The reason for locating the RPV l ft from the Containment centerline is the possibility that the manufacturing tolerances of the circular bridge crane, the location of the rails on the bridge, and the manner in which the main hook cables come off their drums could result in a crane hook travel path that is a chord rather than a diameter of the Containment. If this chord is rotated 360 degrees, a circle is defined at the center of the Containment that is unreachable by the main hook. Were the reactor center to be located inside this circle, the replacement of the RPV head would require a lateral movement to set it in its correct position for closure. By placing the reactor off center, there is a precise hook location at which the reactor head can be set in place without requiring lateral movement.

The lower portion of the primary shield wall provides support for the RPV. A description of the Reactor Vessel Support System is provided in Section 3.8.3.1.8. The primary shield wall provides missile protection and biological shielding and also serves as a support for pipe-whip restraints. Under seismic loading, the primary shield walls serve to provide seismic shear resistance and transmit loading from the upper internals down to the base mat. The bottom of the primary shield wall is anchored into the Containment base slab as shown on Figure 3.8.3-3.

3.8.3.1.2 Secondary Shield Walls: The 3-foot-6-inch-thick secondary shield walls form the exterior of the primary loop compartment. The primary loop compartment is 82 ft wide and 97 ft long and extends from El. (-)11 ft-3 in. to El. 83 ft. The primary shield and refueling pool walls form the interior boundary. The bottom of the compartment is formed by the interior fill slab, while the top is open to the Containment atmosphere. An individual compartment to enclose the pressurizer is provided between SGs no. 1 and no. 4.

As part of steam generator replacement activities, a portion of "D" secondary shield wall in each unit, measuring approximately 20 foot long by 14 foot high, is made removable by cutting a block from the top of the wall and re-attaching it to the adjoining walls using steel splice plates and through bolts. The functions and size of the wall remain unchanged.

The secondary shield walls provide radiation shielding, isolate the RCS, laterally restrain the SGs, RCPs, and pressurizer, support the various piping, serve as pipe-whip restraint supports, and safeguard the electrical and mechanical systems.

3.8.3.1.3 Refueling Cavity: The refueling cavity is a reinforced concrete structure about 21 ft wide by 75 ft long, consisting of the reactor cavity surrounding the upper portion of the RPV and the refueling canal, which connects the fuel storage area and the fuel transfer penetration to the reactor cavity. The reactor cavity and the refueling canal are separated by a stainless steel, manually operated, double-bulkhead gate. The refueling cavity walls are 3 ft-6 in. thick and are lined with stainless steel plate.

The refueling cavity is used during refueling operations to provide shielded access for transferring the new and spent fuel elements between the RPV and the fuel transfer penetration. The reactor cavity is filled with borated water to El. 66 ft 6 in. during those brief periods when a fuel assembly is being transferred over the RPV flange. The refueling cavity also serves as a shielded laydown area for the RPV upper and lower internals.

3.8.3.1.4 Operating Floor: The operating floor at El. 68 ft covers the space between the secondary shield walls and the Containment wall. The floor slab is supported by the secondary shield walls and by beams and columns. A 2-in. gap is left between the Containment wall and the edges of the operating floor and the intermediate floors below to ensure that the only interaction between the Containment wall and the internal structure is through the common foundation base mat.

The function of the operating floor is to provide a working and access floor during refueling, maintenance, and repair operations.

3.8.3.1.5 Intermediate Floors: Intermediate floors between the secondary shield walls and the Containment wall are provided at the following El.: (-)2 ft, 19 ft, 37 ft-3 in., and 52 ft. These floors are supported by structural steel framing spanning between the secondary shield walls and columns and extending up from the base slab at El. (-)11 ft-3 inch. Various access, maintenance and in-service inspection platforms are also provided around equipment.

3.8.3.1.6 Interior Fill Slab: The interior fill slab is 24 in. thick and is placed on top of the foundation mat liner plate. This slab provides protection for the foundation mat liner from any missiles generated in the primary loop compartments and from the effects of temperatures induced by a DBA. Reinforcement is provided to resist temperature and shrinkage forces.

3.8.3.1.7 Polar Crane: A polar crane consisting of a 310-ton main hoist, and a 15-ton auxiliary hoist supported on twin bridge girders is provided inside the RCB for use during construction, maintenance, and repair operations.

The crane moves on a circular rail, which in turn is supported on girders. Brackets anchored on the cylindrical wall through the liner support these girders (Figure 3.8.1-6). The polar crane is anchored to the rails with mechanical guides to prevent its derailment when subjected to earthquake forces.

The polar crane bridge has a rated capacity of 500-tons in Unit 2 and 417-tons in Unit 1. The design of the polar crane trolley assumes it to be loaded with its maximum operating load of 310 tons under both OBE and SSE. The bridge design conservatively assumes the crane to be loaded to 352 tons under both OBE and SSE. Girders and brackets supporting the polar crane are designed to the same loading combinations as the crane bridge.

3.8.3.1.8 Reactor Coolant System Component Supports: The support structures are of welded and/or bolted steel construction of linear and plate types. These supports are tension and compression struts or beams and columns. The supports permit unrestrained thermal growth of the supported system but restrain vertical, lateral, and rotational movement resulting from seismic and pipe-break loadings. This is accomplished using pin-ended columns for vertical supports and girders, hydraulic snubbers, and tie rods for lateral supports.

Shimming and grouting enable adjustment of all support elements during erection to achieve correct fitup and alignment. Final setting of equipment is achieved by shimming and grouting at the building structure/support interface.

3.8.3.1.8.1 Reactor Vessel Supports - The reactor vessel supports consist of individual air-cooled, plate-type support pads as shown on Figure 3.8.3-1. One pad is placed under four of the vessel nozzles and is supported by an embedded plate-type structure which distributes loads to the primary shield wall. Two additional embedded plate type supports transfer lateral forces to the concrete.

In addition to transferring loads from the vessel to the supporting structure, the pads also provide for the passage for cooling through the support to prevent excessive primary shield wall concrete temperatures.

The original design basis postulated pipe break locations in the RCL are described in Reference 3.6-1. The primary RCL components and supports design were based on these postulated break locations. A detailed fracture mechanics evaluation, as described in References 3.6-14, and 3.6-21 through 3.6-29, demonstrates that the probability of rupturing the RCL piping, pressurizer surge line, and the three SIS accumulator lines is extremely low under design basis conditions. Therefore, postulated ruptures in the RCL piping, pressurizer surge line, and the three SIS accumulator lines, and the following associated dynamic effects are not included in the design basis: missile generation, pipe whip, break reaction forces, jet impingement forces, decompression waves within the ruptured pipe, and pressurization in cavities, subcompartments and compartments. The dynamic effects from ruptures in Class 1 branch lines not covered by LBB and other high energy piping are reviewed to verify that the effects are bounded by the current analyses. The seal plates located at the upper reactor cavity are used to provide shielding from neutron and gamma streaming.

The blowdown analysis which determines the adequacy of the reactor vessel supports is comprehensive in that it includes the effects of the hydraulic forces in the loop piping.

3.8.3.1.8.2 Steam Generator - The vertical supports for the SG (Figure 3.8.3-4) consist of four vertical columns bolted at top to the vendor-supplied columns and at bottom to the floor slab. The lower lateral supports consist of supports attached to the walls of each SG subcompartment and bolted to the vendor-supplied beams. The upper lateral supports consist of supports attached to the walls of each SG subcompartment and bolted to the vendor-supplied ring girder around the generator shell connected to hydraulic snubbers and supported by struts on the compartment walls. Loads are transferred from the equipment to the ring girder by means of a number of bumper blocks between the girder and generator shell.

3.8.3.1.8.3 Reactor Coolant Pump - The RCP vertical supports consist of three vertical columns (Figure 3.8.3-5) bolted at top to the vendor-supplied columns and at bottom to the floor slab. The lateral supports consist of three supports attached to the compartment walls and bolted to the vendor-supplied tie-rod supports.

3.8.3.1.8.4 Pressurizer - The pressurizer (Figure 3.8.3-6) is supported at its base by bolting the flange ring to the supporting floor slab. In addition, four lateral supports are provided which are attached to the compartment walls and bolted to the vendor-supplied supports which bear against the vessel lugs.

3.8.3.2 Applicable Codes, Standards and Specifications.

3.8.3.2.1 Codes, Specifications and Standards: The following codes, standards, and specifications are used as a basis for the design, fabrication, construction, testing, and surveillance of the Containment internal structure. Different issue dates of these documents may be used provided they meet the minimum requirements stated herein.

1. American Concrete Institute

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 304-73	-	"Recommended Practice for Measuring, Mining, Transporting and Placing Concrete"
ACI 305-72	-	"Recommended Practice for Hot-Weather Concreting"
ACI 306-72	-	"Recommended Practice for Cold-Weather Concreting"
ACI 308-71	-	"Recommended Practice for Curing Concrete"
ACI 309-72	-	"Recommended Practice for Consolidation of Concrete"
ACI 315-74	-	"Manual of Standard Practice for Detailing Reinforced Concrete Structures"
ACI 318-71	-	"Building Code Requirements for Reinforced Concrete" Exception is taken to Section 9.3, "Required Strength". Refer to Section 3.8.3.3 for loads and loading combinations used in the design of the internal structure.
ACI 347-68	-	"Recommended Practice for Concrete Formwork"
ACI 359-73	-	"Proposed Standard Code for Concrete Reactor Vessels and Containments", ACI 359-ASME, Section III, Division 2, issued for trial use and comment in 1973, including subsequent addenda 1 through 6. Exceptions to the ACI 359 Code are described in Section 3.8.1.6.3.

2. American Institute of Steel Construction

AISC-1969 - "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings", including supplements l, 2, and 3

AISC-1976 - "Specifications for Structural Joints Using ASTM A325 or A490 Bolts"

AISC-1972 - "Code of Standard Practice for Steel Buildings and Bridges"

AISC-1971 - "Structural Steel Detailing"

3. American Welding Society

AWS D1.1-1975 - "Structural Welding Code and Addenda". Visual inspection acceptance criteria for welding in conformance with AWS D1.1 are specifically defined in Appendix 3.8.B. The criteria are incorporated in construction specifications where field welding per AWS D1.1 is specified.

AWS D1.1-1977 - "Structural Welding Code" is used for the pipe whip restraints. Visual inspection acceptance criteria for welding in conformance with AWS D1.1 are specifically defined in Appendix 3.8.B. The criteria are incorporated in construction specifications where field welding per AWS D1.1 is specified.

4. American National Standards Institute

ANSI A58.1-1972 - "American Standard Building Code Requirements for Minimum Design Loads in Buildings and Other Structures"

ANSI N45.2.5-1974 - "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel during the Construction Phase of Nuclear Power Plants"

5. Army Corps of Engineers

CRD-C39 - "Coefficient of Thermal Expansion"

CRD-C44 - "Coefficient of Thermal Conductivity"

CRD-C621 - "Standard Specification for Packaged Dry, Hydraulic-Cement Grout (Nonshrink)"

6. Crane Manufacturers Association of America (CMAA)

CMAA Specification 70

7. American Society of Mechanical Engineers

ASME B&PV Code, Section III, Subsections NA, NE and NF, 1974 edition, including winter 1975 addenda and Code Cases 1644-5, 1644-9 (N71-9), N71-10, 1741, and N-182.

The following exceptions to the code are taken: Code NPT stamping requirements, as per NA-8200 and stress report as per NA-3352 for Steam Generator and Reactor Coolant Pump column supports between El. (-)11 ft-3 in. to El. 16 ft- in., are deleted. These column supports are classified as Category I structural steel but designed and fabricated as per ASME Code. The design of the Fuel Transfer Tube Sleeve system is in accordance with ASME NE code requirements, except that no code stamping will be required.

8. American Society for Testing and Materials (ASTM)

ASTM standards are as referenced herein. Different issue dates of ASTM standards may be used, provided they meet the minimum technical requirements as stated herein.

3.8.3.2.2 Government Regulations and Regulatory Guides: The design, construction, materials, testing, examination, etc., of the Containment internal structures are in conformance with the applicable regulatory guides as listed below and as noted in Section 3.12:

RG 1.10 - "Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures", Exceptions to this guide are stated in Section 3.8.1.6.3.

RG 1.15 - "Testing of Reinforcing Bars for Category I Concrete Structures".

RG 1.55 - "Concrete Placement in Category I Structures".

RG 1.69 - "Concrete Radiation Shields for Nuclear Power Plants".

RG 1.94 - "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants".

3.8.3.3 Loads and Loading Combinations.

3.8.3.3.1 Definitions of Loads: The following nomenclature and definitions apply to all the loads to be encountered and/or to be postulated in the design of the Containment internal structures.

1. Dead Loads (D)

Dead load of the structure plus specific superimposed permanent loads, including the weight and operating loads of major equipment.

Hydrostatic loads and crane loads (without lifted load) are also treated as dead load.

Superimposed and/or suspended loads which account for piping, cable trays, ductwork and miscellaneous equipment distributed throughout floors, are permanent live loads which are considered as equivalent dead loads. All of the dead load components are considered at full value in all loading combinations, including the seismic loading combinations.

2. Live Loads (L)

Floor occupany loads which account for movable equipment, personnel and maintenance loads including construction loads, are designated as temporary occupancy live loads. Laydown area loads are designated as permanent live loads. The live load components designated as temporary occupancy live loads, as defined above, are subject to a 0.25 reduction factor only when considered in the seismic loading combinations. The live load components designated as permanent live loads, as defined above, are considered at full value in all loading combinations, including the seismic loading combinations.

3. DBA Pressure Loads (P_a)

The equivalent static design pressure loadings within or across a compartment occurring as a result of a DBA or a rupture of high-energy line.

4. Operating Thermal Loads (T_o)

Thermal effects on structures based on the most critical steady-state or transient condition during normal operation or shutdown.

5. Operating Piping Loads (R_o)

Piping thrust and thermal expansion forces and reactions based on the most critical steady-state or transient condition during normal operation or shutdown.

6. DBA Thermal Loads (T_a)

Additional thermal effects on structures, above normal operating loads, resulting from a DBA or a rupture of high energy line.

7. OBE Loads (E_o)

Loads generated from the OBE.

8. SSE Loads (E_{ss})

Loads generated from the SSE.

9. DBA Thermal Piping Loads (R_a)

Additional pipe reactions and forces, above normal operating loads, due to thermal effects occurring as a result of a DBA or a rupture of high-energy line.

l0. Pipe Rupture Loads (Y)

Pipe reactions which account for the dynamic effects resulting from postulated rupture of a high-energy pipe. Also included in this rupture loading are direct jet impingement pressure and missile impact effects generated by or during the postulated break. Although the dynamic effects of postulated pipe breaks in the reactor coolant loop primary piping, pressurizer surge line, and SIS accumulator lines can be eliminated from the structural design basis (see Section 3.6.2.1.1.1.a), the design verification of certain structures and components may retain the original pipe break loading.

3.8.3.3.2 Load Combinations: The design of the Containment internal structures, (except RCS equipment supports, control rod drive mechanism, [CRDM] lock lugs and embeds, residual heat removal [RHR] pump supports, RHR heat exchanger [HX] supports, and fuel transfer tube supports which are covered in Section 3.8.3.3.2.3) incorporates two general loading categories:

the Service Load Category and the Nonservice Load Category. Each of these categories is divided into several conditions of loading, which are further subdivided into several different load combinations, as described below.

3.8.3.3.2.1 Service Load Category - This category includes all loading conditions encountered during the construction, normal operation, and shutdown periods of the nuclear power plant. The probability of occurrence of these loads is 1.

A summary of the load combinations for the Service Load Category is shown in Tables 3.8.3-1 and 3.8.3-2 for concrete and steel internal structures, respectively. The concrete and steel internal structures are analyzed and designed to meet the strength requirements for the Service Load Category in accordance with the structural acceptance criteria stipulated in Section 3.8.3.5.

3.8.3.3.2.2 Nonservice Load Category - This category includes all loading conditions resulting from a system failure and/or those extreme environmental conditions postulated to occur during the life of the plant. Also included in this category is the Severe Environmental Condition. The loads in these conditions occur infrequently in combination with normal operating loads. The design probability of occurrence of some of the infrequent loads, such as the OBE, is 1 during the life of the plant, while that of other extreme loads, such as the SSE, is much less than 1.

1. Severe Environmental Condition

   This condition considers all the normal operating loads on the internal structures in combination with the loads resulting from an environmental event, such as the OBE, which may occur only infrequently.

2. Abnormal Condition

   This condition includes the pressure and temperature effects resulting from the DBA. It may also include pipe rupture loads and direct pressure or jet impingement loads generated by a postulated high-energy pipe break accident.

3. Abnormal/Severe Environmental

This condition includes highly infrequent simultaneous occurrence of abnormal and severe environmental effects.

4. Extreme Environmental Condition

This condition includes loads resulting from environmental events which are credible but are highly improbable, such as the SSE.

5. Abnormal/Extreme Environmental Condition

This condition includes the highly improbable simultaneous occurrence of abnormal and extreme environmental effects.

A summary of the load combinations for the Nonservice Load Category is shown in Tables 3.8.3-1 and 3.8.3-2 for concrete and steel internal structures, respectively. The concrete and steel internal structures are analyzed and designed to meet the strength requirements for the Nonservice Load Category in accordance with the structural acceptance criteria stipulated in Section 3.8.3.5.

3.8.3.3.2.3 Reactor Coolant System Support Load Combinations - Steel linear supports for the reactor vessel, SGs, RCPs, pressurizer, RHR pumps, RHR HXs, CRDM anchor lugs, and fuel transfer tube supports are governed by Subsection NF of the ASME Code, Section III, Division 1. These supports are designed for three conditions: normal operating, upset, and faulted. A summary of these load conditions is shown in Table 3.8.3-3. The RCS supports are analyzed and designed in accordance with the elastic method of Paragraph NF-3231.1 of Subsection NF of the ASME B&PV Code, Section III.

3.8.3.3.3 Explanation for Load Factors:

3.8.3.3.3.1 Concrete Structures -

1. Load Factors for the Service Load Category

The load factors for the Service Load Category are conventional and are in accordance with the strength design method of the ACI 318-71 Code.

2. Load Factors for the Nonservice Load Category

a. Severe Environmental Condition

The load factors for the Severe Environmental Condition are based on the strength design method of the ACI 318-71 Code without the standard 75 percent reduction factor for infrequent loadings.

b. Abnormal, Abnormal/Severe Environmental, Extreme Environmental and Abnormal/Extreme Environmental Conditions

A load factor of 1.0 is chosen for the dead loads (D), live loads (L), operating thermal loads (T_o), and the operating piping loads (R_o) because these loads are accurately computable and are associated with an extreme set of conditions not likely to occur.

A load factor of 1.5 is used for the DBA pressure loads (P_a) in the Abnormal Condition.

A load factor of 1.0 is selected for the accident thermal loads (T_a) because these loads result from an extreme set of conditions which has a very remote probability of occurrence.

For the SSE load (E_{ss}), the magnitude of acceleration chosen is representative of the most severe ground motion which can be postulated for this particular site. The intention of utilizing such a load is to demonstrate the functional capability of the structure. Therefore, a load factor of 1.0 is chosen to meet this criteria.

A load factor of 1.0 is used for the pipe rupture loads (Y) because these loads affect only local areas and the intent is to demonstrate that no gross failure of these local areas occurs.

3.8.3.3.3.2 Steel Structures - The design of steel structures is based on Part I of the AISC Specification. Hence, a load factor of 1.0 is used.

3.8.3.3.3.3 RCS Supports - The design of the linear supports for RCS equipment is based on elastic analysis methods of Paragraph NF-3231.1 of Subsection NF of the ASME Code, Section III, Division 1. Hence, a load factor of 1.0 is used.

3.8.3.3.4 Miscellaneous Considerations:

1. For loads which are interrelated as a function of time, such as accident-induced pressure and jet and thermal effects, the maximum values of these effects do not necessarily occur simultaneously. Consideration will be given to the time dependency associated with these postulated failure conditions.

2. The live loads used in the design for each loading combination are assumed consistent with the conditions for that particular combination. Also, live load components are not used to reduce the effects of other applicable loads.

3. The design loading combinations utilized to examine the effects on localized areas, such as loads transferred from support structures, are the same loading combinations utilized for the general internal structure, as described above in Section 3.8.3.3.2.

4. Time-dependent effects, such as creep, shrinkage and other related effects, are included with dead load effects as described in Section 9.3.7 of the ACI 318-71 Code if such loads are of significance in the design of the internal structures.

3.8.3.4 Design and Analysis Procedures.

3.8.3.4.1 Analysis of Concrete Internal Structures: The concrete internal structures are analyzed for all load combinations described in Table 3.8.3-1. Methods of analysis used are based on accepted principles of structural mechanics and are consistent with the geometry and boundary conditions of the structure.

3.8.3.4.1.1 Primary Shield Wall Analysis and Design - The analysis of the primary shield wall is performed by using the BSAP computer program with a three-dimensional finite element model, which consists of brick and boundary elements. The mathematical model is divided into nine layers of brick elements between El. (-)13 ft-3 in. (top of the mat) and El. 38 ft-6-1/2 in. (top of the primary shield wall). Each layer consists of two to six brick elements representing the variable thickness of the wall. A fully fixed boundary condition is assumed at the junction of the primary shield wall with the basemat. Boundary elements are introduced at the appropriate locations to represent the stiffness provided by the secondary shield wall. In addition, nodal loads are applied to the top of the analytical model to account for the effect of the refueling cavity wall due to dead, live and seismic loads.

The loads and loading combination considered for the analysis and design of the primary shield wall are described in Section 3.8.3.3 and Table 3.8.3-1, respectively. The design of reinforcement for the primary shield wall is in accordance with the ACI 318-71 code, and is accomplished by using the OPTCON module of the BSAP-POST.

3.8.3.4.1.2 Secondary Shield Wall Analysis and Design - Secondary shield walls are analyzed using the BSAP Computer program. A three dimensional finite element model is developed to represent the stiffness of the internal structure realistically. The model includes the primary shield wall, an intermediate floor slab, and the principal compartments inside the RCB. The model is a combination of plate, beam, and some boundary elements. The secondary shield walls are assumed hinged at their base whereas the primary shield is assumed fixed.

Loadings considered in the analysis consist of dead, live, and equipment support loads, the three components of earthquake, pipe rupture and jet impingement forces, thermal loads and accident pressure. Dead, live, and equipment support loads are applied to the appropriate elements. The remaining loads are applied to maximize the stresses in the Loop No. 4 compartment. For purposes of analysis, accident pressure is converted to an equivalent static load by applying a dynamic load factor to the calculated peak subcompartment pressures given in Section 6.2.1.2.3.

The analysis results are then used to design the secondary shield wall utilizing the BSAP-POST OPTCON module. Concrete is assumed cracked whenever tensile stresses are present.

3.8.3.4.1.3 Other Concrete Internal Structures - Miscellaneous equipment, compartment slabs, and walls are analyzed using conventional beam/slab design assumptions and equations.

Loadings for these structures consist of dead, live, seismic, pipe rupture, jet impingement, and subcompartment differential pressures where applicable.

3.8.3.4.1.4 Dynamic Analysis Procedures - Earthquake forces on the concrete internal structures are determined by a dynamic analysis in accordance with the techniques described in Section 3.7. The dynamic loads thus determined are then applied as static loads on the concrete structures, and a static analysis using the procedures described above is performed.

The impact effect of the pipe rupture on the structural system is considered by either a conservative energy balance method or by an exact nonlinear time-history analysis. The structural system allowable ductility factors are listed in Table 3.5-13.

For impulse effects such as jet impingement forces, the structural system is allowed to respond inelastically with allowable ductility factors equal to the values listed in Table 3.5-13.

3.8.3.4.2 Analysis of Steel Internal Structures: The steel internal structures are analyzed for all combinations of both service loads and nonservice loads as described in Table 3.8.3-2.

1. Static Analysis Procedures

   The steel internal structures are analyzed for static loads as appropriate either by conventional methods which are well documented in applicable textbooks, or by the Bechtel Structural Analysis Program (BSAP). (See Appendix 3.8.A for a detailed description of the computer programs.)

2. Dynamic Analysis Procedures

   Modal response spectra (MRS) analyses of the integrated floor systems were used for the analysis of seismic loads for design of beams and connections for the internal structural steel.

3. Dynamic effect of pipe rupture is discussed in Section 3.8.3.4.1.4.

3.8.3.4.3 Design and Analysis Procedure for RCS Supports: The linear support systems for components for the SGs, RCPs, and pressurizers are designed by elastic method of analysis. They are analyzed for and designed to resist various combinations of loadings as shown in Table 3.8.3-3. The analysis and design of supports are in accordance with Subsection NF, including Appendix F, Appendix XVII, and Code Class l644-5, ASME B&PV Code, Section III, Division I.

3.8.3.4.4 Design Procedures for Concrete Internal Structures: The concrete internal structures are designed by the strength design method of the ACI 318-71 Code in accordance with the structural acceptance criteria stipulated in Section 3.8.3.5.1. Special considerations in the design of the concrete structures are described below.

1. Geometry of Reinforcing Steel

   In general, all walls and slabs are reinforced in two perpendicular directions at each face. Shear reinforcement is provided as required. Beams and girders are conventionally reinforced using top and bottom longitudinal bars and vertical stirrups. The majority of the bars are of no. 11 size or smaller, thus permitting the use of lapped splices. In the areas where no. 14 or no. 18 size bars are required, Cadweld splices are used.

At the base of the secondary shield walls, primary shield walls, and equipment compartment walls, the vertical reinforcing bars are anchored into the foundation mat. These bars are Cadwelded into each end of a Cadweld sleeve which has been welded into a thickened portion of the liner plate. This arrangement permits the seismic shear to be transferred from the interior structure to the foundation mat. (Refer to Figure 3.8.3-3 for details.)

2. Proportioning of Reinforcing Steel

The results of the analyses under all loading combinations include the moments, axial forces, and shears at each section of the walls, slabs, beams, and columns. Sufficient reinforcing steel is provided to resist the most critical moments, axial forces, and shears as required to satisfy the requirements of ACI 318-71.

3. Bond and Anchorage Requirements of Reinforcing Steel

The provisions of Chapter 12 of the ACI 318-71 Code, "Development of Reinforcement", are complied with in determining bond and anchorage requirements.

3.8.3.4.5 Design Procedures for Steel Internal Structures: The methods of designing the components of the steel internal structures, including design for bending moments, tension and compression forces, connections, and buckling criteria, are in accordance with the procedures outlined in Part I of the AISC Specification, using the structural acceptance criteria stipulated in Section 3.8.3.5.2 for both the service load and the nonservice load combinations. The steel structures are designed in such a way as to behave elastically under all load combinations, with the exception of localized areas subject to missile impact, pipe whip, DBA pressure loads, or SSE forces.

3.8.3.4.6 General Considerations:

1. Design Variables

The general analysis of the concrete internal structures assumes a linear elastic response with uncracked concrete section properties. However, the effects of cracked section properties are considered in the design of critical sections where significant cracking due to thermal stresses is expected. Properties of the reinforced-concrete materials are known with sufficient accuracy and the assumptions made are sufficiently conservative so that other variables need not be considered in the design of the concrete internal structures. For the steel internal structures the use of accepted methods of analysis and the compliance with the AISC Code requirements in the design generally preclude any design variables which may influence the analysis and design results.

2. Interaction with NSSS Equipment Supports

The seismic dynamic analysis of the major NSSS equipment supports (RPV, SGs, RCPs, and pressurizer) considers the interaction between the equipment, the supports, and the concrete and steel components of the internal structure. The internal structures supporting the equipment are designed for the resulting seismic loads transmitted by the equipment supports. Loads transmitted from the equipment supports to the internal structures due to dead weight of equipment, loss of coolant accident, and thermal loads are also considered in the design.

3. Lateral Load Transfer at Foundation Mat

Lateral loads, such as seismic forces or LOCA forces, are transmitted down to the foundation mat primarily by means of "dowel" action through the primary and secondary shield walls. At the base of these walls, the lateral loads are transferred into the foundation base mat through wall dowels.

4. Evaluation of Radiation-Generated Heat Effects

Concrete temperatures do not exceed the values indicated in the ASME-ACI 359 document, Section CC-3430(a), for long-term loading, and Section CC-3430(b), for accident or other short-term loading. If required, insulation and/or cooling systems are provided to limit the temperatures of the concrete to an acceptable level.

3.8.3.5 Structural Acceptance Criteria. The design criteria for the Containment internal structures relating to stresses, strain, gross deformation, factor of safety, and other parameters that identify quantitatively the margin of safety have been briefly discussed in Section 3.8.3.4. In this section the fundamental structural acceptance criteria for the components of the internal structures are listed in greater detail.

3.8.3.5.1 Concrete Internal Structures: The structural acceptance criteria for the concrete internal structures are based on the provisions of the ACI 318-71 Code. The criteria for the concrete structures, as demonstrated by the design calculations, considers service and nonservice load conditions. In order to keep the structural components basically elastic under service load conditions and within the range of general yield, with limited deformations, under nonservice load conditions, the allowable stresses and strains for the strength design method as specified in the ACI 318-71 Code are used; and these allowables are not exceeded when the concrete internal structures are subjected to the loading combinations given in Table 3.8.3-1. The strength capacity of the structure reduced by a capacity reduction factor is equal to or greater than the required strength derived from the loading combinations given in Table 3.8.3-1. Capacity reduction factors (ϕ) as defined in the ACI 318-71 Code are as follows:

1. ϕ = 0.90 for flexure, with or without axial tension

2. ϕ = 0.90 for axial tension

3. ϕ = 0.85 for shear and torsion

4. ϕ = 0.75 for spirally reinforced concrete compression members

5. ϕ = 0.70 for other members in compression

3.8.3.5.1.1 Shear Response of Internal Structures - The concrete shear capacity, including shear reinforcement where required, is in accordance with the requirement of the ACI 318-71 Code.

3.8.3.5.2 Steel Internal Structures: For load combinations l and 3 in Table 3.8.3-2, the allowable stresses for structural steel are in accordance with Part I of AISC specifications. If thermal stresses due to T_o and R_o are present, the allowable stresses for load combinations 2 and 4 are increased by 33 percent.

The permitted increase in allowable stresses for the rest of the combinations are shown in Table 3.8.3-2.

Governing stress ratios for several steel members are provided in Table 3.8.3-4.

NOTE: As indicated in ST-HL-AE-1162 and ST-HL-AE-1250, the information in Table 3.8.3-4 was provided to the NRC as part of the structural design audit conducted in January, 1985. The stress values available at the time of the audit are not necessarily final design values. The values presented in the table do not replace the allowable stress commitments specified in Table 3.8.3-2, and do not represent maximum allowable stresses. (Example: a beam identified in the table as having a stress of 72% of the allowable, has a committed stress value of the allowable, not 72% of the allowable.) The stress values reported in the table provide historical information that remains representative of stress values and governing load combinations existing throughout the building but does not necessarily reflect the largest stresses or the most recent information as documented in the latest design calculations. This note is also applicable to Table 3.8.1-7B.

3.8.3.5.3 Earthquake Response of Interior Structure Related to the Requirement of Attached Equipment: The seismic dynamic analysis of the NSSS equipment considers the interaction between the equipment and its supports, and the internal structure. Deformations of the structures and equipment are checked and, if required, are limited to ensure that no loss of function of any component can occur.

3.83.5.4 Functionality: The Containment internal structures are designed in accordance with the concept of functionality, which defines the behavior for structures which are not defined by strength or stress limitations. Functionality includes such items as preventing excessive deflections or distortions in structural elements, limiting concrete crack size, providing materials for which the structural properties are not degraded by the effects of radiation, etc.

3.8.3.5.5 RCS Support Structures: The RCS component supports are designed to perform within the elastic range for the Normal Operating Condition and Upset Condition in accordance with the limits of Article XVII - 2000 of Appendix XVII of the ASME B&PV Code, Section III, Division I, while the Faulted Condition is designed within general yield strength range at a given temperature in accordance with the limits of F-1370 of Appendix F of the above code.

3.8.3.6 Materials, Quality Control and Special Construction Techniques. Basic construction materials are discussed in this section. Other materials may be used as required by the project specifications. Except as noted in Sections 3.8.3.6.1, 3.8.3.6.2, and 3.8.3.6.3, the materials, quality control, and special construction techniques for the concrete internal structures are as discussed in Section 3.8.1.6.

3.8.3.6.1 Concrete: Concrete for the internal concrete structures is in accordance with Section 3.8.1.6.1, with the exception that the concrete for the primary shield wall and slabs supported by structural steel have a minimum compressive strength of 5,500 psi at 90 days (Class A) and the concrete for all other portions of the internal structure has a minimum compressive strength of 4,000 psi at 28 days (Class B).

3.8.3.6.2 Reinforcing Steel: The requirements for reinforcing steel are the same as those of Section 3.8.1.6.2 except as noted below:

Sub-Subparagraphs CC-4333.4.2, Splice Samples, and CC-4333.4.4, Tensile Testing Requirements, serve as an alternate to Section C.3 of RG 1.10 except that the location of all cadweld splices, including replacement splices, is maintained by pour number or by as-built sketches and additional records are maintained showing the location and test results of all splice samples tested. These records are in addition to the requirements of Subsubparagraph CC-4333.1.2, Maintenance and Certification of Records.

3.8.3.6.3 Structural Steel:

3.8.3.6.3.1 Materials - Basic materials used in the structural and miscellaneous steel construction conform to the following ASTM standards. Additional standard specifications or different issue dates of the standards may be used provided they meet the minimum technical requirements as stated herein. Fuel pool liner plate gate and fuel transfer tube sleeve materials are included.

ASTM A168	Carbon steel rails
ASTM A36-75	Rolled shapes, plates, and bars
ASTM A53-73	Steel pipe
ASTM A106-77	Steel pipe
ASTM A108-73	Weld studs
ASTM A123-73	Zinc coatings (hot galvanized)
ASTM A153-73	Zinc coating on hardware
ASTM A193-74	Bolting material
ASTM A194-75	Carbon and Alloy Steel nuts
ASTM A234-77a	Pipe fittings
ASTM A240-75, type 304	Stainless steel plate
ASTM A276-75	Stainless and heat-resisting steel bars and plates
ASTM A283-70	Low and Intermediate Tensile strength steel plates
ASTM A285-74a	Pressure Vessel Plates (CST)
ASTM A307-68	Low-carbon steel bolts NOTE: The 5/16-inch-diameter bolts and nuts used for companion flange connections of HVAC ductworks may depart from the thread tolerances specified by the ANSI B1.1 standards included in ASTM A307. The departure in thread fit is due to the allowed intermixing of galvanized and electroplated bolting material.
ASTM A312-76	Stainless steel pipe
ASTM A325-76	High-strength bolts
ASTM A358-75, type 304	Electric-fusion-welded austenitic chromium-nickel alloy steel pipe
ASTM A441-70	High-strength steel
ASTM A449-78	Quenched and tempered steel bolts and studs

ASTM A479-76		Stainless steel bars and shapes
ASTM A480-72		Stainless steel plate, sheet, strip
ASTM A490-71		Quenched and tempered steel bolts
ASTM A501-74		Hot-formed welded and seamless steel tubing
ASTM A519-76		Seamless carbon and alloy steel mech. tubing
ASTM A540-75		Alloy steel bolting materials
ASTM A570-75		Carbon hot-rolled sheet and strip
ASTM A588-71		High-strength, low-alloy steel
ASTM A618-74		Welded and seamless high-strength low-alloy tubing
ASTM A668-79		Steel forgings
ASME SA-240,	Gr. 304	Heat-resisting chromium and chromium-nickel stainless steel plate
ASME SA-358,	Gr. 304	Electric-fusion-welded austenitic chromium-nickel alloy steel pipe
ASME SA-516,	Gr.70	Heat-resisting chromium and chromium-nickel stainless steel plate

3.8.3.6.3.2 Fabrication and Erection of Structural Steel - The fabrication and erection of structural steel is in compliance with the requirements of the AISC Specification.

3.8.3.6.3.3 Quality Control Procedures - Structural and miscellaneous steel is examined and tested in accordance with the AISC Specification and the material purchase specifications. The certification of steel material is documented either by Certified Material Test Reports (CMTR) that define the applicable chemical and physical properties or by Certificates of Conformance (C of C) that assert compliance with the prescribed material specification. The required form of material certification is defined in the project specifications. Material marking and identification are in accordance with the material purchase specifications. Shop and field structural welding is performed by qualified welders in accordance with approved welding procedures. Documentation of the welder qualification test, the properties of the welding electrodes used and the procedures for NDE is maintained by the user for audit and/or surveillance by the owner or his representative.

The QA program for the construction phase is described in the Quality Assurance Program Description. The QA program for the operations phase is described in the Operations Quality Assurance Plan.

3.8.3.6.4 RCS Supports:

3.8.3.6.4.1 Materials - ASTM A588 corrosion-resistant, high-strength, low-alloy steel for structural shapes, plates, and bars is used. ASTM A540 alloy steel bolting material, Grade B-23, Class 4 is used for bolts and nuts. Quenched and tempered carbon steel (0.40 percent minimum carbon) or ASTM A325 material is used for washers. The design values of yield strength, S_y, at design temperature and minimum ultimate tensile strength, S_u, for A588 and A540 are used in accordance with Subsection NF, including Appendix F, Appendix XVII, and Code Case l644-5 and 1644-9 (N71-9) of the ASME B&PV Code, Section III, Division I.

3.8.3.6.4.2 Quality Control Procedures - The QC program for the fabrication, NDE of the materials to determine physical properties, and construction, including erection tolerances, are in accordance with ANSI N45.2 and Subsection NF of the ASME B&PV Code, Section III, Division I.

3.8.3.7 Testing and Inservice Surveillance Requirements.

3.8.3.7.1 Concrete Internal Structures: No testing, such as differential pressure testing of individual compartments, or inservice surveillance is required for the concrete internal structures other than the pneumatic pressure test of the entire Containment interior, as described in Section 3.8.1.7. Compliance with the testing and QC procedures described in Sections 3.8.3.6 and 3.8.1.6.4.2 assures the structural integrity of the concrete internal structures.

3.8.3.7.2 Steel Internal Structures: Except for the testing and QC practices described in Section 3.8.3.6, no further tests or inservice surveillance is required for the structural steel and connections. The structural steel framing and connections are generally accessible to visual inspection; and, if necessary, these elements are inspected by NDT procedures after the occurrence of any major catastrophic phenomenon, such as an SSE or a missile impact, to ensure their structural integrity.

3.8.3.7.3 Stainless Steel Liner for Refueling Canal: A Leak Collection System is provided behind the seam welds of the stainless steel liner of the refueling canal for periodic inservice monitoring for leaks.

3.8.3.7.4 Polar Crane and Jib Crane Testing: The polar crane is tested to 125 percent of its rated capacity, in accordance with the requirements of ANSI B30.2.0 (referenced in CMAA specification 70). In accordance with the requirements of ASME B30.4, the jib cranes are tested to 125 percent at their rated capacity and the jib crane platforms are tested to 110 percent of the rated load in the direction of the highest generated moments.

3.8.3.7.5 RCS Supports: The RCS component supports testing and inservice surveillance requirements are in accordance with Articles NF-4000 and NF-5000 of the NF Subsection of the ASME B&PV Code, Section III, Division I.

3.8.4 Other Category I Structures

The following seismic Category I structures are described in this section:

1. Mechanical-Electrical Auxiliaries Building (MEAB)

2. Diesel Generator Building (DGB)

3. Fuel Handling Building (FHB)

4. Essential Cooling Water Intake Structure (ECWIS)

5. Essential Cooling Water Discharge Structure

6. Class 1E Underground Electrical Raceway System

7. Auxiliary Feedwater Storage Tank (AFST)

Figure 3.8.4-1 shows the arrangement of these structures and also illustrates which of the facilities are common to the two units and which are separately provided for each unit. For further details, refer to Section 1.2. All Category I structures have been designed to resist the loads generated by OBE, DBA, SSE, pipe rupture, missiles, wind, tornado, and flood, except otherwise noted.

3.8.4.1 Description of Structures. Category I structures are independently supported and sufficiently separated from adjoining structures to prevent interaction due to building settlement, thermal expansion, and lateral deflections due to seismic effects. The separation of 3 in. minimum between adjacent Category I structures is provided by seismic joint material which allows free movement and rotation between buildings.

3.8.4.1.1 Mechanical-Electrical Auxiliaries Building: This building is a multistoried structure which houses mechanical equipment, electrical equipment, and the isolation valve cubicle. These three areas are separated by reinforced concrete walls and supported on a common foundation mat.

The mechanical section of the building (called the Mechanical Auxiliary Building [MAB]) measures 245 ft long by 199 ft wide for Unit 1 and 244 ft long by 199 ft wide for Unit 2. The highest portion of the roof is at El. 95 and the mat is 18 ft below grade (plant grade is El. 28 ft). This section houses and supports the ESF systems, waste processing systems, piping systems, and the auxiliary equipment. In addition, there is a 7.5-ton overhead bridge crane necessary for handling radioactive solid waste. The interior arrangement of reinforced-concrete columns, walls, and slabs in the building reflects a separation of systems concepts by utilizing pipe chases, valve galleries and isolated equipment compartments. This interior arrangement is designed to facilitate optimization of systems, reduce radioactive contamination of clean areas from potentially contaminated areas, and provide maximum radiation protection to personnel.

The electrical section of the building (called the Electrical Auxiliary Building [EAB]) measures 254 ft long by 123 ft wide for Unit 1 and 253 ft long by 123 ft wide for Unit 2, with the highest portion of the roof at El. 96 ft and mat 18 ft below grade. This section houses and supports the Class 1E electrical controls, switchgear, battery room, computer room and cable raceways. The floors and the roof are supported by structural steel beams, girders, columns, and reinforced-concrete walls. In the EAB, general physical separation between the three redundant ESF trains is attained by locating each train on one of three different elevations enclosed by fire-rated floors, walls, and ceilings. The control room is a column-free area, and the cable spread area has split-level floors to utilize the space more effectively. Special fire protection systems have been utilized for all the exposed structural steel members throughout the entire building.

The isolation valve cubicles section of the building measures 82 ft long by 62 ft wide. The highest portions of the split-level roof are at El. 95 ft, and the top of basemat is 18 ft below grade. Walls extend continuously to El. 86 ft and El. 95 ft-5 in. to provide fire and environmental separation for the four cubicles. A sheet metal roof is provided to protect the cubicle from inclement weather, but which will blow off during a tornado or postulated break in the main steam, feedwater, or other high pressure lines. Loss of the roof during these postulated events limits the pressure in the cubicles.

A probabilistic risk assessment of the isolation valve cubicles without tornado barriers at the roof results in less than 10^{-7} probability of tornado missile strike on safety-related equipment.

The physical arrangement of all three sections for both units are identical except for the following two items:

- The distance between column lines H and J is 20 ft 0 in. for Unit 1 and 19 ft-0 in. for Unit 2.

- The thickness of the concrete wall along column line M_8 between column lines 28 and 32 is 3 ft-0 in. for Unit 1 and 2 ft-6 in. for Unit 2.

3.8.4.1.2 Diesel Generator Building: This two-story building measures 107 ft long by 82 ft wide, with the roof at El. 82 ft, 100 ft, and 107 ft, and mat 3 ft below grade. This building houses and supports three DGs, three 75,000-gallon diesel oil tanks, and three pieces of DG intake and exhaust equipment. Each DG, with its diesel oil storage tank and piece of intake and exhaust equipment, is located in a separate compartment divided from the others by reinforced-concrete slabs and walls. The exterior walls and roof slab are designed to prevent tornado-missile penetration, except at the DG exhaust penetrations (see Section 9.5.8.2 for details).

3.8.4.1.3 Fuel Handling Building: The FHB is a reinforced concrete structure 185 ft long, 87 ft-6 in. wide, and 93 ft above grade (El. 28 ft). The foundation mat is placed at four different levels: railroad track area mat and new fuel-receiving area slab 2 ft above grade; cask decontamination area mat at 4 ft below grade; Liquid Waste Processing System (LWPS) surge tank and HVAC Supply and Exhaust Subsystem area mat 24 ft below grade; and the spent fuel pool slab 6 ft below grade, under which safety injection pumps area mat is 57 ft below grade. The FHB houses new fuel, spent fuel, fuel shipping container and cask, spent fuel pool HX, spent fuel pool pumps, skimmer pumps, low-head and high-head safety injection pumps, Containment spray pumps, and valve isolation tank.

The building enclosure above and below the operating level (El. 68 ft) consists of reinforced-concrete walls and roof slab capable of resisting tornado-generated missiles. The wall thickness is designed to be at least equal to the missile penetration depth. The spent fuel cask transfer area is separated from the spent fuel pool by a reinforced-concrete wall. The 150-ton cask handling crane is physically prevented from operating over the spent fuel pool, thus eliminating any possibility of its accidentally falling into the pool. A 15/2-ton crane is provided for equipment handling. The new fuel elements are handled by a separate 5-ton bridge crane. The handling of spent fuel in the spent fuel pool is done by a fuel handling machine, which runs on rails along the top of spent fuel pool walls. The spent fuel is transferred from RCB to spent fuel pool via transfer tube and transfer canal. The spent fuel pool and fuel transfer canals are lined with stainless steel plate with a leak detection system behind the liner to ensure leaktight integrity. When completed, the cask loading pool will have similar construction.

3.8.4.1.4 Essential Cooling Water Intake Structure: The ECWIS, a shared facility between Unit 1 and Unit 2, is a reinforced-concrete building. The building is located at the Essential Cooling Pond (ECP). The structure is divided into six compartments, three for each unit. The dividing walls between the essential cooling water (ECW) pumps, as well as the external walls, are designed to protect the individual pumps and motors to preclude the loss of independence.

Each compartment houses the ECW pump and strainer and is equipped with trash rack, traveling screen, and screen wash pump. The structure measures approximately 136 ft long, 79 ft-6 in. wide, 21 ft above grade (El. 34 ft), and the bottom of the sump mat is 24 ft below grade.

3.8.4.1.5 Essential Cooling Water Discharge Structure: The ECW Discharge Structure, a shared facility between Unit 1 and Unit 2, is a reinforced-concrete building. The building is located adjacent to the ECWIS. The structure is divided into two compartments, one for each unit. Each compartment houses the end portion of the return pipelines. The water in the pond (from El. 17 ft to 26 ft) provides a cushioning effect for the falling water from the return lines (Centerline El. 31 ft-9 in.). In front of the structures towards the ECP, a reinforced-concrete apron is provided to preclude erosion of soil in the vicinity of the structure. The structure measures approximately 53 ft long by 48 ft wide. The foundation mat is El. 19 ft and the roof of the structure is El. 36 ft.

3.8.4.1.6 Class 1E Underground Electrical Raceway System: The Class 1E Underground Electrical Raceway System provides electrical distribution from the MEAB to the DGB, the ECWIS, and the TGB. The raceway system consists of banks of PVC conduits in a spaced arrangement encased in reinforced concrete.

The devised structural system is capable of supporting its own weight and other external loads. Manholes are provided at specified intervals along the system. Ductbanks are a minimum 4 ft below the finished grade level except between the MEAB and TGB where they are generally at grade and slope 3 in. per l00 ft toward the manholes.

3.8.4.1.7 Auxiliary Feedwater Storage Tank: The AFST is a reinforced-concrete structure with cylindrical walls covered by a circular slab. The tank measures approximately 50 ft in diameter and 47 ft high. The tank is supported on a circular concrete mat. The inside of the tank has a stainless steel liner. The tank has a usable 525,000-gallon demineralized water storage capacity. The exterior wall and roof slab are designed to prevent tornado-missile penetrations.

3.8.4.2 Applicable Codes, Standards, and Specifications.

3.8.4.2.1 Codes, Standards, and Specifications: The following codes, standards and specifications are used as a basis for the design, fabrication, construction, testing, and surveillance of other Category I Structures. Different issue dates of the documents may be used provided they meet the minimum technical requirements stated herein.

1. Uniform Building Code (UBC) - 1973

2. ANSI A58.1-1972, "American National Standard Building Code Requirements for Minimum Design Loads in Buildings and Other Structures"

3. ACI 318-1971, "Building Code Requirements for Reinforced Concrete"

4. ACI 336-1972 "Suggested Design Procedures for Combined Footings and Mats"

5. ACI 347-1968 "Recommended Practice for Concrete Formwork"

6. AISC – Manual of Steel Construction, Seventh Edition

7. AISC 1972 "Code of Standard Practice for Steel Buildings and Bridges"

8. National Fire Protection Association (NFPA) 1973, Codes and Standards

9. American Welding Society D1.1-75, "AWS Structural Welding Code". Visual inspection acceptance criteria for welding in conformance with AWS D1.1 are specifically defined in Appendix 3.8.B. The criteria are incorporated in construction specifications where welding per AWS D1.1 is specified.

10. ACI – Manual of Concrete Practice (Part I & II - 1973, Part III - 1972)

11. AISC 1969 "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings", including supplements 1, 2, and 3.

12. ASME, Section VIII, Division 1, 1974 including Winter 1975 addenda.

13. ASME, Section IX, Division 1, 1974 including Winter 1975 addenda (for Fuel Transfer Tube bellows only)

14. ASME, Section IX, Division 1, 1971 including Winter 1973 addenda

15. ASME, Section III, Division 1, 1974 including Winter 1975 addenda

16. ASME, Section II, Part II, Part C, 1974 including Winter 1975 addenda

17. Army Corps of Engineers – Handbook of Concrete and Cement

18. CMAA Specification 70 (1971)

19. ACI 315, 1974 – Manual of Standard Practice for Detailing Reinforced Concrete Structures

20. AISC-1976 - "Specification for Structural (Joints Using ASTM A325 or A490 Bolts"

21. AISC, 1971 - "Structural Steel Detailing" (1971)

22. ANSI B18.3-1976 - "Socket Cap, Shoulder and Set Screws - Inch Series"

3.8.4.2.2 Government Regulations and Regulatory Guides: The design, construction, materials, testing, examination, etc., of the other Category I structures are in conformance with government regulations as discussed in Section 3.1 and with the following RGs as stated in Section 3.12:

RG 1.10 "Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures". An exception is taken to RG 1.10, as previously stated in Sections 3.8.1.6.3 and 3.8.3.6.2.

RG 1.13 "Spent Fuel Storage Facility Design Basis"

RG 1.15 "Testing of Reinforcing Bars for Category I Concrete Structures"

RG 1.29 "Seismic Design Classification"

RG 1.55 "Concrete Placement in Category I Structures"

RG 1.59 "Design Basis Floods for Nuclear Power Plants"

RG 1.64 "Quality Assurance Requirements for the Design of Nuclear Power Plants"

RG 1.69 "Concrete Radiation Shields"

RG 1.76 "Design Basis Tornado"

The following guides are not applicable to STPEGS per the implementation portion of the guide; however, degree of compliance is addressed in the UFSAR.

RG 1.94 "Q.A. Requirement for Installation, Inspection and Testing Structural Concrete and Structural Steel During Construction"

RG 1.102 "Flood Protection for Nuclear Power Plants"

3.8.4.3 Loads and Loading Combinations.

3.8.4.3.1 Definitions of Loads: The following nomenclature and definitions apply to the loads to be encountered and/or to be postulated in the design of the Category I structures other than Containment.

1. Dead Loads (D)

Dead load of the structure plus specific superimposed permanent loads, including the weight and operating loads of major equipment. Hydrostatic and crane loads (without lifted loads) are also treated as dead load.

Superimposed and suspended loads which account for piping, cable trays, ductwork and miscellaneous equipment distributed throughout floors, are permanent live loads which are considered as equivalent dead loads. All of the dead load components are considered at full value in all loading combinations, including the seismic loading combinations.

2. Live Loads (L)

Floor occupancy loads which account for movable equipment, personnel and maintenance loads, including construction loads, are designated as temporary occupancy live loads. Laydown area loads and soil pressure load are designated as permanent live load. The live load components designated as temporary occupancy live loads, as defined above, are subject to a 0.25 reduction factor only when considered in the seismic loading combinations. The live load components designated as permanent live loads, as defined above, are considered at full value in all loading combinations, including the seismic loading combinations.

3. Operating Thermal Loads (T_o)

Thermal effect on structures based on the most critical steady-state or transient condition during normal operation.

4. Operating Piping Loads (R_o)

Piping thrust and thermal expansion forces and reactions based on the most critical steady-state or transient condition during normal operation.

5. OBE Loads (E_o)

Loads generated from OBE.

6. Wind Loads (W)

Loads generated by the design basis wind. Refer to Section 3.3.

7. SSE Loads (E_{ss})

Loads generated from SSE. Refer to Section 3.7.

8. Tornado Loads (W_t)

Loads generated by the design tornado. Also included is the simultaneous occurrence of a given pressure increase within a particular structure and the effects of tornado-generated missiles. Refer to Sections 3.3 and 3.5.

9. DBA Pressure Loads (P_a)

The equivalent static design pressure which would result from DBA pipe rupture.

10. DBA Thermal Load (T_a)

Additional thermal effects above normal operating loads on a structure which would result from DBA pipe rupture.

11. Accident Thermal Pipe Loads (R_a)

Pipe reactions due to thermal effects generated by the postulated break of a high-energy pipe.

12. Pipe Rupture Loads (Y)

Pipe reactions which account for the dynamic effects resulting from a postulated rupture of a high-energy pipe. Also included in this rupture loading are direct jet impingement pressure and missile impact effects generated by or during the postulated break.

In determining an appropriate equivalent static load for Y, elastoplastic behavior has been assumed with appropriate ductility ratios, provided excessive deflections will not result in loss of function of any safety-related system.

13. Flood Loads (H)

Hydrostatic and buoyancy forces are due to a failure of the reservoir embankment. Also included are hydrodynamic effects due to wave action. Refer to Section 3.4. These loads are in addition to normal groundwater loads. See Load no. 1 in Section 3.8.4.3.1.

3.8.4.3.2 Load Combinations: Category I structures other than Containment are subjected to two types of load categories, each consisting of several conditions of loading, which are further subdivided into several different loading combinations.

3.8.4.3.2.1 Service Load Category - The Service Load Category includes all loading conditions encountered throughout the construction and normal condition of the nuclear power plant including severe environmental loads. The design probability of occurrence of these loads is l during the life of the plant.

1. Normal Condition

This condition includes all loads which are expected to be encountered during normal plant operation, test startup, refueling, and shutdown. This condition includes all loads applied to a structure during its construction period and which would affect the structural integrity of the building during its design life.

2. Severe Environmental Condition

Loads included in this condition are from environmental events that occur only infrequently during the plant life. These events include the OBE and the design wind.

A summary of the service load combinations is shown in Tables 3.8.4-1 and 3.8.4-2.

3.8.4.3.2.2 Nonservice Load Category - The Nonservice Load Category includes all loading conditions resulting from a system failure and/or those extreme environmental conditions postulated to occur during the life of the plant. The probability of extreme loads such as tornado and SSE is much less than 1.

1. Abnormal Condition

This condition includes pressure and temperature effects resulting from the DBA.

2. Extreme Environmental Condition

This condition includes loads resulting from environmental events which are credible but are highly improbable. These events include flood, the SSE, and the design tornado.

3. Abnormal/Extreme Environmental Condition

This condition includes the highly improbable simultaneous occurrence of abnormal and extreme environmental effects.

A summary of the nonservice load combinations is shown in Tables 3.8.4.-1 and 3.8.4-2.

3.8.4.3.3 Explanation for Load Factors:

3.8.4.3.3.1 Concrete Structures -

1. Load Factors Under the Service Load Category

   a. The load factors under the Service Load Category are conventional and are based on the strength design method of the ACI 318-71 Code. This also satisfies the requirements of Sections 9.3.4 and 9.3.5 of ACI 318-71 where soil and hydrostatic pressure are present as L and D, respectively.

   b. The load factors under the Severe Environmental Condition are based on the strength design method of the ACI 318-71 Code without the standard 75 percent reduction factor for wind and OBE loading. This also satisfies requirements of Sections 9.3.4 and 9.3.5 of ACI 318-71 where soil and hydrostatic pressure are present as L and D, respectively.

2. Load Factors Under the Nonservice Load Category

   a. The load factors under the Abnormal, Extreme Environmental, and Abnormal/Extreme Environmental Conditions shall be as follows:

      1) Dead Load, Operating Thermal Loads, Operating Piping Loads, and Live Loads (D, T_o, R_o, L)

   These loads are accurately computable and are associated with an extreme set of conditions which are not liable to occur. Therefore, a load factor of 1.0 is used.

      2) Safe Shutdown Earthquake Loads (E_{ss})

         The magnitude of acceleration chosen is representative of the most severe ground motion which could be postulated for this particular site. The intention of utilizing such a load is to demonstrate the functional capability of the structure and, therefore, a load factor of 1.0 was chosen to meet this criteria.

      3) Pipe Rupture Loads, Tornado Loads, Flood Loads, DBA Thermal Loads (Y, W_t, H, T_a)

         A load factor of 1.0 is used with each of these loads because of their highly remote occurrence.

3.8.4.3.3.2 Steel Structures - The design of steel structures is based on Part I of AISC Design Specification; hence, a load factor of 1.0 is used.

3.8.4.3.4 Explanation For Load Combinations: All load combinations in Tables 3.8.4-1 and 3.8.4-2 are based on standard design codes and are consistent with current NRC positions.

3.8.4.4 Design and Analysis Procedures. The design and analysis procedures for all other Category I structures, including assumptions on boundary conditions and expected behavior under loads, are in accordance with ACI 318-71 for concrete structures and with AISC Manual of Steel Construction for steel structures. The design and analysis of structures based on load combinations are given in Section 3.8.4.3 and the "Method for Missile Analysis" in Section 3.5.3. Earthquake forces on the structures are determined by a dynamic analysis (Section 3.7) and then applied statically in the design of structures.

1. Computer Programs

The structural analysis is performed by the computer program BSAP.

2. Expansion-Type Seal Joints

Buildings which are separated by a 3-in. expansion gap are connected by watertight joints with waterstop. The seal joint is sufficiently flexible to absorb the movement between structures without exceeding the safety limits.

3. Concrete Masonry Unit (CMU) Walls

Presently, there are no safety-related CMU walls inside any of the seismic Category I structures. If safety-related CMU walls are deemed necessary in the future to enhance plant function, such walls will not serve as major load bearing elements and will not be used as part of the overall building shear wall system.

When such CMU walls are used, materials, testing, analysis, design, construction and inspection requirements will conform to the applicable requirements of the UBC-1979 except as noted below:

a. Load Combinations:

1) Service Load Conditions

a) $D + L$

b) $D + L + E_o$

c) $D + L + W$

If thermal stresses due to T_o and R_o are present, they should be included in the above combinations, as follows:

1 - $D + L + T_o + R_o$

2 - $D + L + T_o + R_o + E_o$

3 - $D + L + T_o + R_o + W$

Check load combination for controlling condition for maximum 'L' and for no 'L'.

2) Extreme Environmental, Abnormal, Abnormal/Severe Environmental, and Abnormal/Extreme Environmental Conditions

a) $D + L + T_o + R_o + E_{ss}$

b) $D + L + T_o + R_o + W_t$

c) $D + L + T_a + R_a + 1.5 P_a + T_o$

d) $D + L + T_a + R_a + 1.25 P_a + 1.0 Y + 1.25 E_o + T_o$

e) $D + L + T_a + R_a + 1.0\ P_a + 1.0\ Y + 1.0\ E_{ss} + T_o$

In combinations 2)c), 2)d), and 2)e), the maximum values of P_a, T_a, R_a, and Y, including an appropriate dynamic load factor, should be used unless a time-history analysis is performed to justify otherwise. Combinations 2)b), 2)d), and 2)e) and the corresponding structural acceptance criteria is satisfied first without the tornado missile load in 2)b) and without Y in 2)d) and 2)e). When considering these loads, local section strength capacities may be exceeded under these concentrated loads, provided there will be no loss of function of any safety-related system.

Both cases of L having its full value or being completely absent should be checked.

b. Allowable Stresses

Allowable stresses provided in ACI-531-79, as supplemented by the following modifications/exceptions, shall apply.

1) When wind or seismic loads (OBE) are considered in the loading combinations, no increase in the allowable stresses is permitted.

2) Use of allowable stresses corresponding to special inspection category shall be substantiated by demonstration of compliance with the inspection requirements of the NRC criteria.

3) In qualifying masonry walls, no credit will be taken for tension perpendicular to bed joints. All the tensile stresses will be resisted by reinforcement.

4) For load conditions which represent extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions, the allowable working stress may be multiplied by the factors shown in the following table:

Type of Stress	Factor
Axial or Flexural Compression[1]	2.5
Bearing	2.5
Reinforcement stress except shear	2.0 but not to exceed 0.9 fy
Shear reinforcement and/or bolts	1.5
Masonry tension parallel to bed joint	1.5

[1] When anchor bolts are used, design should prevent facial spalling of masonry unit.

Shear carried by masonry	1.3
Masonry tension perpendicular to bed joint for reinforced masonry	0

Safety-related CMU walls are currently not planned inside Category I structures, however, if CMU walls are necessary within a Category I structure and are in proximity to safety-related systems and equipment such that wall failure could adversely affect a safety-related system or equipment they shall be designed as safety-related CMU walls.

Removable CMU walls, which are built with masonry or concrete units stacked without any grouting or reinforcing, will be restrained to prevent collapse of the units onto safety-related equipment. The restraint system will consist of steel framing provided on both faces of the walls.

No safety-related piping systems or equipment are attached to the CMU walls.

3.8.4.4.1 Mechanical-Electrical Auxiliaries Building:

For the gravity loads, the Mechanical Auxiliary Building has been analyzed by utilizing a combination of conventional type analysis (columns and beams), and finite element modeling (for large slabs) incorporating the designated shear walls.

The analysis for the EAB was based on use of conventional steel framing supported on steel columns and bearing walls. The major structural steel beams have been designed utilizing the composite action between steel and concrete.

The analysis of the isolation valve cubicle is a combination of conventional-type analysis and finite element modeling considering the effect of main steam and feedwater pipe rupture loads and other applicable loads.

For the lateral loads the entire building is considered to be comprised of a system of floors and walls acting as horizontal diaphragms and shear walls, respectively. The lateral loads are assumed to be concentrated at the floor levels and are distributed to exterior walls and selected interior walls in accordance with their continuity between floor levels and in proportion to their stiffnesses. Exterior walls are shear walls designed to sustain wind, tornadoes, and seismic loads, to act as missile barriers, and to provide radioactive shielding for outside atmosphere. All interior walls are designed to act as bearing and in selected cases as shear walls, and are designed to sustain seismic loads and to provide radioactive shielding and fire protection barrier where it is required.

In certain local areas such as large openings or plate elements subject to high concentrated loads which cannot be analyzed within the conventional, practical limits, finite element models are used, and the effect of boundary conditions at the interconnections of the structural members is determined according to their relative stiffnesses.

The finite element model is also used when the geometry of the walls makes it difficult to model the wall in a conventional way.

3.8.4.4.2 Diesel Generator Building: The reinforced-concrete roof and intermediate slabs are designed for dead, live, tornado, seismic and thermal loads according to load combinations given in Table 3.8.4-1. The exterior walls are designed for flood and tornado loads. The interior and exterior walls around the diesel oil tanks have a 3-hour fire rating. The slab supporting the tank is designed for the static load and the seismic overturning moment from the tank.

The DGs are supported on the mat resting on soil. The mat is also designed for static and dynamic loads transferred by walls, in addition to equipment forces.

3.8.4.4.3 Fuel Handling Building: The roof slab is designed as a one-way slab with composite beams supported by roof trusses. The supporting structural steel is designed in accordance with the AISC manual. The roof slab, intermediate floor slabs, and exterior and interior walls are designed for applicable combinations of tornado, missile, flood, seismic, dead, and live loads in accordance with the ultimate strength design method of ACI 318-71.

The spent fuel pool is designed as an open tank to carry hydrostatic, hydrodynamic, and thermal loads. The hydrodynamic forces due to earthquake are calculated based on TID-7024, "Nuclear Reactor and Earthquake". The thermal gradients have been used to design for thermal loads. In the transfer canal, the transfer tube is fitted with sleeve and expansion bellows to absorb relative movement between the RCB, Containment internals, and the FHB. The bellows design considered all loading conditions, including SSE and maximum hydraulic pressure. A connecting channel is located between the spent fuel pool and the cask loading pool. The slabs forming the bottom of the decontamination area and the bottom of cask loading pool are analyzed to satisfy the requirements of SRP Section 15.7.5. The spent fuel pool and fuel transfer canals are lined with stainless steel plate with a leak detection system behind the liner to ensure leaktight integrity. When completed, the cask loading pool will have similar construction.

The new fuel storage pit has a three-section hatch cover. The slab at the bottom of the pit is designed to withstand the impact force of a new fuel assembly dropped from maximum elevation allowed by the 2-ton hoist of the fuel handling overhead crane.

Corbels supporting the 150-ton overhead crane, 15/2-ton crane, 5-ton new fuel crane and the spent fuel pool fuel handling machine are designed to withstand lateral and axial loads of the SSE and OBE.

The exterior subsurface walls are designed as rigid restrained walls to resist combined axial and lateral static, at-rest, and dynamic pressure under seismic conditions.

3.8.4.4.4 Essential Cooling Water Intake Structure: The ECWIS roof slab and walls are designed for dead, live load, tornado and seismic. The walls are designed for hydrostatic and hydrodynamic loads as per TID 7024, "Nuclear Reactor and Earthquake." The walls of the structure towards the pond are designed to resist wave impact forces.

3.8.4.4.5 Essential Cooling Water Discharge Structure: The ECW Discharge Structure is divided into two layers, one for each unit. The structure is reinforced-concrete abutment with flared-wing walls such that eddies do not continue beyond wing walls and is designed for lateral soil pressure based on ultimate strength design method of ACI 318-71. The end portion of the discharge piping is protected by slab and wall, which is designed for tornado missiles. In front of the structure and towards the ECP, a reinforced concrete apron is provided to prevent erosion of soil in the vicinity of the structure. The discharge piping line enters the ECP above normal water level so that a pipe rupture cannot siphon any of the water supply from the pond. Falling water has sufficient height above water level to provide a cushioning effect.

3.8.4.4.6 Class 1E Underground Electrical Raceway System: In designing the Class 1E Underground Electrical Raceway System the following loads are considered:

1. Dead load (D)

2. Live load (L) due to surcharge, crane and/or railroad

3. Seismic load (E_o, E_{ss})

4. Missile due to tornado

5. Hydrostatic (flood) load (H)

6. Thermal load (T_o)

7. Differential settlement

The raceway system consists of two independent structures, namely, ductbank and manhole. At the junction of the two, there is a 2-in. flexible joint so movement of one structure in any direction will not affect the other (total relative movement less than 2 in.).

The first five loads, above, are considered for manhole design, while all the loads are considered for ductbank design.

1. Manhole Design

Each component of manhole (e.g., top and bottom slabs, walls, etc.) has been designed discretely with appropriate load combinations and boundary conditions resulting from continuity. The top slab has been designed for support of its own deadweight and the soil above it. Lateral earth pressure and hydrostatic pressure on walls have been taken as dead load on walls. Surcharge loads and crane loads (construction load) have been considered as live loads. For seismic loads, maximum ground accelerations of 0.05g for OBE and 0.10g for SSE have been taken (Section 3.7.1). Increase in lateral earth pressure due to seismic effect has also been considered. The 2 ft thickness of the top slab was determined from a missile of 1-inch-diameter steel rod, 3 ft long, weighing 8 pounds, and striking vertically at a speed of 310 ft/sec. Any loss of velocity while passing through the soil over the slab is ignored. The worst-flood condition (23 ft of water above grade level) is also considered.

All possible load combinations are taken into account in accordance with Tables 3.8.4-1 and 3.8.4-2.

Finally, concrete sections and reinforcements are designed in accordance with ACI 318-1971 (Section 3.8.4.2.1).

2. Ductbank Design

a. Ductbanks have been designed as beams on elastic foundation. Since the dead load (own weight and soil load) is uniform, no significant stress is induced in the ductbanks. Surcharge load is also uniform and thus induces no stress. Crane load and railroad load have been considered. A tornado generated missile has been considered to determine the concrete cover to protect the cable ducts. During operation, the cables will slowly heat up to an operating temperature. There will be a temperature gradient between the outside face of concrete and PVC ducts. Hence, the ductbanks are designed for this temperature gradient.

b. The following procedures have been used to obtain the design loads:

1) Dead Loads (D): Dead load of the system includes weights of conduits, cables, and soil on top of ductbanks and manholes in addition to its own weight. Hydrostatic loads are considered as dead loads.

2) Live Loads (L): Surcharge, lateral soil pressure loads and railroad loads are considered as live loads. During construction, cranes may travel over the ductbanks. Thus, the ductbanks are checked for a crane load of 240 tons.

3) OBE (E_o): During a seismic event, the ductbanks are assumed to develop strains and deformations, same as in the soil media, due to friction between soil and ductbanks. The following are the two types of seismic waves that are considered for the design of ductbanks:

a) Compression waves (P-waves)
b) Shear waves (S-waves)

Stresses in the ductbanks are obtained by multiplying the strains (due to waves) by the modulus of elasticity (E) of respective materials (either concrete or reinforcing steel). Since maximum stresses due to various seismic waves do not occur simultaneously, the representative maximum stress is computed by taking the square root of the sum of the squares of stresses due to all seismic waves. Maximum ground velocity which is also required for computation of stresses due to seismic waves is taken as 48 in./sec for 1g ground acceleration for OBE.

4) SSE (E_{ss}): Maximum horizontal and vertical accelerations are twice those of OBE.

5) Tornado Loads (W_t): No tornado wind load shall be applicable since the ductbank system is an underground structure. Nevertheless, tornado generated missile loads are considered for the design of manholes and ductbanks.

6) Thermal Loads (T_o): Temperature in the ductbanks shall be 75°C at the operating condition. A linear variation of temperature gradient from the conduit to outside face of concrete is assumed. Temperature in the soil shall be 75°F.

7) Differential Settlement (D_s): A differential settlement of 1.5 in. in the ducts (about 50 ft long) adjacent to the MEAB is considered. From this deflection, an equivalent concentrated load is determined assuming that each ductbank is a semiinfinite beam resting on an elastic continuous support having one end free and subjected to a concentrated load at the free end.

c. Load combinations

All the loads (i.e., 1 through 7) are considered in accordance with Table 3.8.4-1. It is to be noted that construction and railroad loads are not combined with seismic or flood loads. The railroad load is combined with differential settlement load only. Also note that the equivalent concentrated load as calculated from differential settlement is added with all other loads with a load factor of 1.0.

d. Design Procedures

Applicable codes, standards and specifications are described in Sections 3.8.4.2.1 and 3.8.4.2.2. Reinforced concrete design is done according to ACI 318-71.

The values of soil parameters used in the analysis are as follows:

1) Coefficient of Subgrade Reaction = K_s

The Category I Electrical Raceway Ducts are located between the Unit 1 MEAB and the ECWIS. The ductbanks are constructed both in the structural backfill and the clay A_2-layer.

The structural design of the electric raceway duct is based on a coefficient of subgrade reaction of 150 kcf in the structural backfill and 15 kcf in the A_2 clay layer.

The 150 kcf is conservatively taken as 0.25 times ksi value of 600 kcf as recommended by Terzaghi (Ref. 2.5.4-47) for dense submerged sand, which is the same derivation as described in Section 2.5.4.10.4.2. The 15 kcf is taken as 0.25 times ksi value of 60 kcf as recommended by Terzaghi (Ref. 2.5.4-47) for stiff clay. This was checked by Vesic's method with an average value of 2.20 ksf for A_2 clays's elastic modulus. This elastic modulus was conservatively taken from results at 0.5 percent strain of laboratory unconsolidated-undrained triaxial tests. These K_s values are applicable to short term loading conditions.

2) Shear Wave Velocity and Poisson's Ratio

For shear wave velocity (V_s) refer to Section 3.7.1.4.3.

A poisson ratio (γ) of 0.42 is selected for the design, which is based on published data by Barkan (Ref. 2.5.4-4) and Leonards (Ref. 2.5.4-26).

e. Materials

Normal weight concrete that is used has a 28-day compressive strength of 4000 psi except where the duck bank from the MEAB to the TGB where 5500 psi concrete is used for tornado missile protection. Reinforcing steel that is used is deformed billet steel bars conforming to ASTM A 615, Grade 60 (f_y = 60 ksi).

Finally, ductbank sections are designed according to ACI 318-1971 (Section 3.8.4.2.1).

3.8.4.4.7 Auxiliary Feedwater Storage Tanks: The roof slab of the AFST is designed for tornado wind, thermal loads and missile penetration. The walls and base slab are designed for hydrostatic, hydrodynamic, thermal tornado wind, missile penetration, and flood loads. The interior surfaces of the tank have a stainless steel liner to ensure leaktight integrity.

The liner is fabricated in accordance with ASME Code Section III, Subsection ND and erected in accordance with AISC specifications, but is not a Class 3 pressure boundary. The tank has a nitrogen blanket of approximately 5 in. of water column.

3.8.4.5 Structural Acceptance Criteria.

3.8.4.5.1 Reinforced Concrete:

1. Stress and Strain Criteria

For all other Category I concrete structures, the allowable stress and strain criteria are in accordance with the strength design method of the ACI 318-71 Building Code.

The margin of structural safety has been provided in two ways. One is in the form of a load factor which is associated with each type of load. The magnitude of the load factors in the load combinations is based on the nature of the applied load, probability of occurrence during the life of the structure, and the accuracy in predicting the magnitude of the load. Second, an additional safety measure has been incorporated by use of a capacity-reduction factor described in Section 9.2 of the ACI 318-71 Building Code. This capacity-reduction factor accounts for variations in material strengths, workmanship and dimensions.

2. Functionality

Functionality defines behavior which is not defined by strength of stress limitations. The minimum requirements for reinforcement, elastic deflection, vibration, and slenderness ratio are in accordance with the ACI 318-71 Building Code. In addition to design requirements, the member sizes will be made similar at some places to satisfy construction requirements and aesthetic aspects.

3.8.4.5.2 Structural Steel: Allowable strength for each combination is specified in Table 3.8.4-2. The design for structural steel is in accordance with Part I of AISC Specification.

1. Missile Barrier

The local and overall effects of missiles on the structural system are considered in accordance with design procedures indicated in Section 3.5.3. Barriers are provided wherever necessary to protect equipment from the effects of missiles, pipe whip, and jet impingement.

2. Pipe Restraints

Protection to structural system and components from the damages resulting from the high-energy pipe rupture is accomplished in some situations by providing restraints at critical locations of piping systems. Stress criteria for pipe restraints are discussed in Section 3.6.

3.8.4.6 Materials, Quality Control (QC) and Special Construction Techniques. Materials, QC and special construction techniques for other Category I structures are as discussed in Section 3.8.3.6.

3.8.4.7 Testing and Inservice Surveillance Requirements.

1. Concrete and Steel Structures

Other than the compliance of the requirements of Section 3.8.4.6 on materials, QC, and construction techniques, there will be no required planned systematic testing or surveillance, except occasional visual inspection.

2. Stainless Steel Liner

The testing for leaktightness of the stainless steel liner for the spent fuel pool will be performed through the Leak Chase System after the pool is filled with water. In addition, the Leak Chase System will be subjected to periodic inservice monitoring for any possible leakages from the spent fuel pool.

3.8.5 Foundations and Concrete Supports

3.8.5.1 Description of the Foundations and Supports. The foundation of Category I structures consists of reinforced-concrete mats supported on undisturbed soil or engineered structural

backfill material. Figure 3.8.5-1 illustrates the location and physical separation of Category I structures.
Seismic interaction between buildings is avoided by providing a 3 in. minimum gap.

To prevent the possibility of groundwater or surface runoff from seeping into buildings, waterproofing membranes are applied to all exterior surfaces below finished-grade level. In addition, waterstops are provided between the mats of Category I structures.

Embedded foundation reinforcing steel and the exterior of the liner plate will be protected from corrosion by concrete. A design measure to help control corrosion will be the application of a waterproofing membrane applied to the external concrete surfaces below grade. These measures will suffice to control the corrosion of rebar and the liner plate in the RCB. The combination of the waterproofing membrane and the remoteness of the liner plate from the soil environment will suffice to control corrosion. In addition, the underground portion of the liner plate and embedded steel are provided with cathodic protection.

The effects of floods on the building foundation have been covered in Section 3.4.

Typical details illustrating methods of anchorage of large equipment and vertical structural elements to foundation are shown on Figures 3.8.3-1 and 3.8.3-3.

A discussion of the effects of dynamic lateral earth pressures on foundations and concrete supports is provided in Appendix 3.7.A.

3.8.5.1.1 Reactor Containment Building: For a description of the RCB foundations and supports, see Section 3.8.1.

3.8.5.1.2 Mechanical-Electrical Auxiliaries Building: The MEAB is founded on a 6-foot-thick reinforced-concrete mat. The top of the mat is 18 ft below grade and is supported on engineered structural backfill.

The mat is designed to transmit all loads from the superstructure's shear-bearing walls and columns to the soil. For out-of-plane loads and axial loads, the superstructure is usually considered pinned to the mat in order to maximize the moment effects in the mat. The superstructure is considered to be rigidly attached to the mat when transmitting in-plane shear loads from shear walls. The foundation mat is modeled by finite elements in areas where columns are the principal vertical load applying members (EAB area and part of the MAB). In other areas, conventional two-way slab analysis techniques are used with walls acting as the reaction edges for the mat. Horizontal forces are resisted by soil/structure interaction. The finite element analysis of the mat is performed using the BSAP computer program.

Major equipment such as tanks and heat exchangers is rigidly connected to the mat by anchor bolts which transmit lateral loads to the foundation mat.

3.8.5.1.3 Diesel Generator Building: The DGB is supported on a mat with exterior and interior bearing walls. The top of the mat is 3 ft below grade. The DGs are supported on the same mat foundation. Forces, reactions and displacements are determined by the finite element method using the BSAP computer program. The mat is considered to be supported in the vertical direction by elastic coil springs.

3.8.5.1.4 Fuel Handling Building: The FHB is supported on base slabs at three different levels: under the spent fuel pool area, 57 ft below grade; under the cask loading area, 24 ft below grade; and at the railroad track, 2 ft above grade. The mat at 57 ft below grade is supported on undisturbed soil, and the other areas are on engineered structural back-fill. The nine caissons for pumps under the lowest mat are designed for hydrostatic, soil and surcharge pressures under normal conditions. Under the Extreme Environment Accident, the mat is designed for maximum passive soil resistance. Also, wing walls are designed to resist sliding under the seismic loads. The mat has been designed according to the loads and load combination of Section 3.8.4.3. The exterior walls which are subjected to lateral loads have been considered in the mat design. The foundation mat model consists of plate finite elements supported on linear elastic-springs. The stiffness of the springs are a function of coefficient of subgrade modulus and surrounding area. Forces, reactions and displacement are found using the computer program BSAP. The major equipment in this building, such as tank, pump, and HX, is rigidly connected to slabs through anchor bolts which transmit the equipment loads and lateral loads to the foundations.

3.8.5.1.5 Essential Cooling Water Intake Structure: The foundation of the ECWIS is supported on engineered structural backfill. The foundations of the ECW Intake and Discharge Structures are physically separated from each other. The design is based on the analytical method as described in ACI 336-72. The reinforcing and stress requirements of vertical structural walls and the base slab and beam-wall joints for structures comply with ACI 318-71, including a special provision for seismic design.

3.8.5.1.6 Essential Cooling Water Discharge Structure: The foundation of the ECW Discharge Structure is supported on engineered structural backfill. The base slab and walls retaining earth pressure are designed for load combinations according to Section 3.8.4.3, and the reinforcing pattern and stress requirements comply with ACI 318-71.

3.8.5.1.7 Class 1E Underground Electrical Raceway System: The foundation of the Class 1E Underground Electrical Raceway System is supported on undisturbed soil and/or engineered structural backfill. For foundation design criteria, see Section 3.8.4.4.6.

3.8.5.1.8 Auxiliary Feedwater Storage Tank: The base slab of the tank is supported on engineered structural backfill. The base slab design (circular in shape and uniform thickness) specifies a slab resting on an elastic foundation for load combination according to Section 3.8.4.3. The stress requirements, due to the constrained condition of vertical walls subjected to hydrostatic, hydrodynamic and temperature loads, are accounted for in the design of the base slab.

3.8.5.2 Applicable Codes, Standards and Specifications. The pertinent codes, standards, specifications, NRC regulations, and RGs governing the design, construction, fabrication, inspection, testing, and material properties for foundations and concrete supports are referenced in the following sections:

Containment Structure	Section 3.8.1.2
Containment Internals	Section 3.8.3.2
Other Category I Structures	Section 3.8.4.2

3.8.5.3 Loads and Loading Combinations.

3.8.5.3.1 General: The loads and loading combinations to which Category I foundations and supports are subjected, as well as load factors for design approach used, are described in the following sections:

Containment Structure	Sections 3.8.1.3 and 3.8.1.4
Containment Internals	Sections 3.8.3.3 and 3.8.3.4
Other Category I Structures	Sections 3.8.4.3 and 3.8.4.4

The load transfer mechanism and various effects such as base shear, torsional moment, pressure redistribution, etc., are discussed in the following sections:

Containment Structure	Section 3.8.1.4
Containment Internals	Section 3.8.3.4
Other Category I Structures	Section 3.8.5.4

3.8.5.3.2 Gross and Differential Settlements: All Category I structures are supported on isolated foundations so that all differential settlements of any one foundation do not impose loads on the adjacent structures. A discussion on the expected settlement of structures is presented in Section 2.5.4.

3.8.5.4 Design and Analysis Procedures. The design and analysis procedures used in designing the structural foundations or elements thereof, and various equipment supports, including the assumptions made and boundary conditions used, are described in the following sections:

Containment	Section 3.8.1.4
Containment Internals	Section 3.8.3.4
Other Category I Structures	Section 3.8.5.4.1

3.8.5.4.1 Other Category I Structures: The foundation of the building is considered to consist of base mat and the lower section of walls which account for the stiffness of the superstructure.

The analysis of the foundation follows this sequence: find the foundation pressure in the subsoil under the base mat; distribute soil reactions; analyze the base mat for internal forces and deformations. The analytical details involved are as follows:

1. Foundation Pressure

The foundation pressures under the base mat are determined for the unfactored loads of Section 3.8.5.3 without considering the superstructure stiffness. The resulting values are compared with the allowable bearing capacity as discussed for various structures in Section 2.5.4. Lateral loads and forces and overturning moments generated by wind, tornadoes, earthquakes, and pipe rupture are transmitted through walls from structures to foundation media. Method of determining overturning moments due to the three components of the earthquake is described in Section 3.7.2.6.

2. Distribution of Soil Reactions

The contact pressure at the base of a foundation has been considered either an elastic subgrade reaction, or a straight line distribution. The foundation pressures between the lease mat and the subsoil are distributed to be compatible with the deflection of the substructure and the soil displacement.

3. Internal Forces and Deformation

To assure proper determination of moment and shear, the base mat is analyzed either by finite elements or by conventional two-way slab analysis depending on the complexity of the column and slab arrangements. The stiffness effects of walls and columns are included in the analysis only if the inclusion increases stresses in the mat. Boundary conditions are considered either pinned or fixed depending on the stiffness of the supporting structure. Where the combination of mat thickness and support spacing permits, rigid body analysis is conservatively performed to calculate soil pressures. These pressures are then used as loads for the mat which is assumed supported in columns and/or walls, as applicable. For other mats, soil springs are used to model the flexibility of the soil.

When finite element analysis is performed, the computer program BSAP is used.

4. Load Transfer

The loads of the superstructure and equipment and the imposed forces are transferred to the foundation mat through the reactions of the structural system. The further transfer of loads from foundation mat to the supporting soil is achieved by direct bearing, surface friction, and lateral passive resistance.

5. Torsional Moments

The effect of torsional moments on the mat foundations caused by the eccentric forces on the superstructure are considered for the loading combinations specified in Section 3.8.4.3.

3.8.5.5 Structural Acceptance Criteria. The structural acceptance criteria relating to stresses, strains and gross deformations of foundation mats of buildings are described in the following sections:

Containment Structure	Section 3.8.1.5
Containment Internals	Section 3.8.3.5
Other Category I Structures	Section 3.8.4.5

Overturning and Sliding of Structures

The following safety factors apply to the load combinations given below for the Containment and other Category I structures.

LOAD COMBINATION	MINIMUM FACTORS OF SAFETY		
	OVERTURNING	SLIDING	FLOTATION
$D + F' + E_O$	1.5	1.5	-
$D + F' + W$	1.5	1.5	-
$D + F' + E_{ss}$	1.1	1.1	-
$D + F' + W_t$	1.1	1.1	-
$D + H$	-	-	1.1

D, E_o, W, E_{ss}, W_a and H are defined in Section 3.8.4 and F' is the lateral earth pressure.

Factors of safety against shear failure in the soil, differential settlements, limiting conditions of stresses, strains and deformations in soil, and other conditions that identify quantitatively the margin of safety against the loading combinations specified for the building are specified in Section 2.5.4.

3.8.5.6 Material Specifications, Quality Control, and Special Construction Techniques. The material specifications, quality control procedures, and special construction techniques used for foundations and supports are the same as those for structures which are supported thereon. They are identified in Section 2.5.4 and in the following respective sections:

Containment Structure	Section 3.8.1.6
Containment Internals	Section 3.8.3.6
Other Category I Structures	Section 3.8.4.6

3.8.5.7 Testing and Inservice Surveillance Requirements. There are no planned systematic testing or inservice surveillance programs, other than a visual inspection after they are completed, for the Category I concrete foundations.

The requirements for inservice surveillance of concrete supports are the same as those for other Category I structures and are identified in the following sections:

Supports Located within Containment	Section 3.8.3.7
Other Category I Supports	Section 3.8.4.7

TABLE 3.8.1-1

LOAD COMBINATIONS FOR CONTAINMENT STRUCTURE

LOADS

CATEGORY	LOADING CONDITION	NO.	D	L	F	T_o	R_o	E_o	W	E_{ss}	W_t	H	T_a	R_a	P_a	T_t	P_v	Y
	CONSTRUCTION	1	1.0	1.0	1.0	1.0												
	TESTS	2	1.0	1.0	1.0										1.15	1.0	1.0	
SERVICE	NORMAL	3	1.0	1.0	1.0	1.0	1.0										1.0	
	SEVERE	4	1.0	1.0	1.0	1.0	1.0	1.0									1.0	
	ENVIRONMENTAL	5	1.0	1.0	1.0	1.0	1.0		1.0								1.0	
	SEVERE	6	1.0	1.0	1.0	1.0	1.0	1.5									1.0	
	ENVIRONMENTAL	7	1.0	1.0	1.3	1.0	1.0		1.5								1.0	
		8	1.0	1.0	1.0	1.0	1.0	1.0 or	1.0									
		9	1.0	1.0	1.0	1.0	1.0			1.0							1.0	
	EXTREME	10	1.0	1.0	1.0	1.0	1.0				1.0						1.0	
	ENVIRONMENTAL	11	1.0	1.0	1.0	1.0	1.0					1.0					1.0	
NON-	ABNORMAL	12	1.0	1.0	1.0	1.0	1.0						1.0	1.0	1.0			
SERVICE		13	1.0	1.0	1.0	1.0	1.25						1.0	1.25	1.0			
	ABNORMAL/SEVERE	14	1.0	1.0	1.0	1.0	1.0	1.25					1.0	1.0	1.25			
	ENVIRONMENTAL	15	1.0	1.0	1.0	1.0	1.0		1.25				1.0	1.0	1.25			
	ABNORMAL/EXTREME ENVIRONMENTAL	16	1.0	1.0	1.0	1.0	1.0			1.0			1.0	1.0	1.0			1.0

In combinations 12 through 16, the maximum values of P_a, $(T_o + T_a)$, $(R_o + R_a)$ and Y including an appropriate dynamic load factor shall be used unless a time history analysis is performed to justify otherwise.

TABLE 3.8.1-2
SUMMARY OF IN-PROCESS TEST RESULTS
PORTLAND CEMENT

Compound/Property	Range Max.	Range Min.	Avg.	Standard Deviation	C/V%	Initial[1]
Autoclave expansion	+0.58	-0.07	-0.01	N/A	N/A	0.02
Initial set	4:30	1:25	2:58	0:36	20.4	2:30
Final set	7:30	2:45	5:10	1:02	20.0	5:20
False set, %	100	55	81	11.9	14.6	97
3 - day strength	2,910	1,040	2,055	331	16.1	2,570
7 - day strength	3,840	1,930	2,848	398	14.0	3,360
28 - day strength	6,090	3,040	4,597	633	13.8	5,900
Air content, %	11.0	4.9	8.4	1.1	12.6	9.2
Blaine	4,411	2,911	3,443	274	8.0	3,284
SiO_2	23.56	21.46	22.37	0.44	2.0	22.82
$A1_2O_3$	5.06	3.05	4.26	0.36	8.3	4.24
Fe_2O_3	5.86	3.32	4.25	0.46	10.9	3.82
MgO	1.32	0.61	0.89	0.13	14.2	0.80
SO_3	2.68	1.57	2.18	0.20	9.3	2.28
Loss on ignition	2.94	0.78	1.79	0.45	25.2	0.94
Insol. Residue	0.60	0.04	0.15	0.07	42.4	0.13
C_3S	55.3	30.7	47.8	4.4	9.2	48.2
C_3A	7.1	0.1	4.1	1.3	32.1	4.7
$C_3S + C_3A$	60.5	39.9	52.0	4.1	7.9	52.9
CaO	64.9	62.0	63.56	0.57	0.9	64.4
Na_2O	0.28	0.00	0.08	0.04	51.2	0.08
K_2O	0.69	0.28	0.46	0.10	21.2	0.48
Total alk. @ Na_2O	0.60	0.25	0.39	0.09	22.4	0.40

Number of tests: 156 (Grinds 105 through 261[2])

1. Preliminary Acceptance Test (Grind No. 103)
2. Grind 109 was rejected and is not included.

TABLE 3.8.1-3A

SUMMARY OF IN-PROCESS TEST RESULTS
SIEVE ANALYSIS AND FINENESS MODULUS
FINE AGGREGATE (SAND)

Sieve Size	Percent Passing Maximum	Minimum	Average	Initial[1]
3/8 in.	100	100	100	100
No. 4	100	98	100	100
No. 8	100	84	96.6	95
No. 16	83	61	74.4	70
No. 30	55	25	41.3	37
No. 50	30	5	14.1	13
No. 100	6	1	2.6	3
No. 200 (wash)	1	0.3	0.5	1
F.M.	3.1	2.5	2.72	2.82

Number of Tests - 2448 (March, 1977 through November, 1981)

1. Preliminary Acceptance Test

TABLE 3.8.1-3B

SUMMARY OF IN-PROCESS TEST RESULTS
FINE AGGREGATE (SAND)

Property	Range Max.	Range Min.	Avg.	Standard Deviation	No. of Tests	Initial[1]
Friable particles (%)	1.0	0.0	0.28	0.23	69	0.00
Lightweight particles (%)	0.4	0.0	0.12	0.11	69	0.10
Absorption (%)	0.9	0.1	0.69	0.10	68	0.90
Specific gravity (SSD)	2.68	2.59	2.62	0.02	68	2.62
Reduction in alkalinity	See	Note 2	--	--	12	64.9
Dissolved silica	See	Note 2	--	--	12	20.00
$MgSO_4$ soundness (%)[4]	6.3	0.4	3.21	N/A[3]	10	2.7
$NaSO_4$ soundness (%)[4]	2.4	1.4	1.90	N/A[3]	2	2.0

Testing Period: April 1976 through December 1981 (69 months)

1. Preliminary Acceptance Test
2. Reduction in alkalinity and dissolved silica determinations must be considered in combination for individual samples. Two of twelve, or 17%, of the semiannual tests were considered as potentially reactive.
3. Insufficient data.
4. Five cycles.

TABLE 3.8.1-3C

SUMMARY OF IN-PROCESS TEST RESULTS
SIEVE ANALYSIS
COARSE AGGREGATES

	PERCENT PASSING			
	Number 4 (1-1/2 in.)		Number 67 (3/4 in.)	
Sieve Size	Average	Initial	Average	Initial[1]
2 in.	100	100	N/A	N/A
1-½ in.	99.7	100	N/A	N/A
1 in.	37.7	51	100	100
3/4 in.	6.7	8	98.6	100
½ in.	Not req'd	3	Not req'd	81
3/8 in.	1.0	2	38.0	39
No. 4	N/A	1	5.9	3
No. 8	N/A	N/A	1.0	1
Number of Tests	870		2350	

Testing Period: (March 1977 through November 1981)

1. Preliminary Acceptance Test

TABLE 3.8.1-3D

SUMMARY OF IN-PROCESS TEST RESULTS
AGGREGATE NO. 67 (3/4 IN. GRAVEL)

Property	Range Max.	Range Min.	Avg.	Standard Deviation	No. of Tests	Initial[1]
Flat and elongated (%)	7.9	0.0	2.78	1.78	69	2.1
Friable particles (%)	2.0	0.0	0.15	0.43	65	0.00
Lightweight particles (%)	0.4	0.0	0.09	0.09	69	0.00
Soft particles (%)	1.0	0.0	0.04	0.21	69	0.80
Absorption (%)	1.3	0.7	1.06	0.13	68	0.90
Specific gravity	2.63	2.55	2.58	0.01	68	2.57
L.A. abrasion	27.0	19.8	22.3	N/A[3]	12	22.2
Reduction in alkalinity	See	Note 2	--	--	12	138.35
Dissolved silica	See	Note 2	--	--	12	214.00
$MgSO_4$ soundness (%)[4]	1.8	0.1	0.89	N/A[3]	10	1.30
$NaSO_4$ soundness (%)[4]	0.9	0.4	0.65	N/A[3]	2	0.50

Testing Period: April 1976 through December 1981 (69 months)

1. Preliminary Acceptance Test
2. Reduction in alkalinity and dissolved silica determinations must be considered in combination for individual samples. Ten of twelve, or 83% of the semiannual tests were considered as potentially reactive.
3. Insufficient data
4. Five cycles

TABLE 3.8.1-3E

SUMMARY OF IN-PROCESS TEST RESULTS
AGGREGATE NO. 4 (1-½ IN. GRAVEL)

Property	Range Max.	Range Min.	Avg.	Standard Deviation	No. of Tests	Initial[1]
Flat and elongated (%)	7.0	0.0	2.36	1.71	69	0.3
Friable particles (%)	2.0	0.0	0.09	0.28	65	0.00
Lightweight particles (%)	0.4	0.0	0.06	0.12	69	0.00
Soft particles (%)	1.0	0.0	0.04	0.18	68	0.70
Absorption (%)	0.8	0.3	0.62	0.08	68	0.60
Specific gravity	2.61	2.52	2.58	0.01	68	2.60
L.A. abrasion (%)	21.6	18.5	19.8	N/A[3]	12	19.6
Reduction in alkalinity	See	Note 2	--	--	12	172.51
Dissolved silica	See	Note 2	--	--	12	400.00
$MgSO_4$ soundness (%)[4]	1.0	0.2	0.39	N/A[3]	10	0.20
$NaSO_4$ soundness (%)[4]	0.4	0.3	0.35	N/A[3]	2	0.30

Testing Period: April 1976 through December 1981 (69 months)

1. Preliminary Acceptance Test
2. Reduction in alkalinity and dissolved silica determinations must be considered in combination for individual samples. Eight of twelve, or 67% of the semiannual tests were considered as potentially reactive.
3. Insufficient data
4. Five cycles

TABLE 3.8.1-4

SUMMARY OF IN-PROCESS TEST RESULTS
MIX WATER

Property	± Variance[1] Range Max.	Min.	Avg.	Number of Tests	± Variance[1] Initial[2]
Initial time of set, vicat (min)	12	0	4.8	66	10
Final time of set, vicat (min)	55	0	13.6	66	5
Autoclave expansion	+0.08	-0.04	0.00	66	0.0
7 - day compressive strength (%)	+15.4	-8.9	+0.3	66	+0.8
28 - day compressive strength (%)	+9.7	-8.5	-1.4	66	-4.9

	Range Max.	Min.	Avg.	Number of Tests	Initial[2]
Chlorides (ppm)	209.9	10.1	92.6	67	95.42
Solids (ppm)	634.0	53.0	474.0	67	581.00
Sulfates (ppm)	38.0	0.1	11.2	59	7.00

Testing Period: February 1976 through December 1981

1. Comparison of test water with control water
2. Preliminary acceptance tests

TABLE 3.8.1-5

SUMMARY OF IN-PROCESS TEST RESULTS
ADMIXTURES[1]

	DAREX AEA				DARATARD HC				WRDA HC			
	Range				Range				Range			
Property	Max.	Min.	Avg.	Initial[2]	Max.	Min.	Avg.	Initial[2]	Max.	Min.	Avg.	Initial[2]
Solids (%)	6.13	5.60	5.92	5.53	35.79	34.36	35.30	38.28	35.94	34.40	35.13	35.87
Specific gravity	1.02	1.01	1.01	1.01	1.20	1.19	1.20	1.21	1.17	1.13	1.14	1.12
pH	8.5	7.0	8.0	6.4	11.5	10.3	10.7	9.3	11.3	9.3	10.8	9.3
Chloride (ppm)	420	80	189	50	475	10	89	290	389	5	143	40
Number of Tests			15				20				8	

Testing Period: March 1976 through December 1981

1. Data from CMTRs
2. Preliminary acceptance tests by testing laboratory

TABLE 3.8.1-6

IN-PROCESS SUMMARY OF FIELD COMPRESSIVE STRENGTHS AND PLASTIC DATA

			Plastic Tests				Compressive Strength Tests											
Mix I.D[(1)]	No. of Tests	Period.	Avg. Temp.	Avg. Slump	Avg. Air Con-tent	Avg. Unit Wt.	7-Day Avg. Str.	W/in Test Var.	Std. Dev.	Overall C of V	28-Day Avg. Str.	W.in Test Var.	Std. Dev.	Overall C of V	90 Day Avg. Str.	W/in Test Str.	Std. Dev.	Overall C of V
A-1-3-01	80	04/23/76-10/19/77	65	3-3/4	4.6	144.1	4790	4.0	381	8.0	6510	4.2	663	10.2	7550	4.2	746	9.9
A-1-3-20	433	04/04/77-11/05/81	64	4-3/4	4.4	144.0	4210	4.0	477	11.3	6130	4.5	634	10.3	7470	3.8	700	9.4
A-2-2-01	10	01/07/77-03/03/77	57	3-1/4	3.5	146.2	4600	-	-	-	6430	-	-	-	7560	-	-	-
A-2-3-01	30	11/11/76-01/05/77	63	3-1/2	3.6	146.3	4380	4.5	322	7.3	5960	3.4	343	5.8	7110	5.3	431	6.1
A-2-3-20	79	05/18/77-11/06/78	62	4-1/4	4.2	145.6	4190	4.4	431	10.3	5840	5.4	537	9.2	7190	4.2	678	9.4
B-1-3-01	208	04/08/76-04/06/77	69	3-1/4	4.0	144.7	4130	3.7	335	8.1	5970	3.9	399	6.7	-	-	-	-
B-1-3-10	142	04/05/77-10/03/77	71	3-1/2	4.2	143.7	3350	3.6	338	10.1	5010	3.8	442	8.8	-	-	-	-
B-1-3-11	2335	10/05/77-11/10/81	64	4-1/4	4.2	143.9	3390	3.2	398	11.7	5440	3.3	468	8.6	-	-	-	-
B-2-2-01	75	01/04/77-02/04/77	57	3-1/4	4.0	145.3	3480	4.4	276	7.9	5450	4.3	276	5.1	-	-	-	-
B-2-3-01	329	10/11/76-04/05/77	63	3-1/4	4.0	146.0	3470	4.1	314	9.0	5200	3.9	433	8.3	-	-	-	-
B-2-3-10	1164	04/06/77-11/04/81	64	3-3/4	4.0	145.6	3390	3.9	351	10.4	5160	4.1	500	9.7	-	-	-	-
C-1-2-21[(2)]	245	10/03/77-11/10/81	65	4-3/4	4.2	142.8	2110	3.5	314	14.9	3870	3.2	427	11.0	4960	2.7	518	10.5
C-2-3-20[(2)]	22	05/17/77-06/17/77	70	3	4.2	143.2	1910	5.2	184	9.6	3240	3.9	213	6.6	4020	3.7	292	7.3
E-2-3-01[(2)]	58	04/09/76-05/02/77	67	3	4.2	144.0	1980	4.3	324	16.4	3430	3.8	390	11.4	-	-	-	-
B-1-3-11[(3)] Special	61	12/02/80-01/20/81	61	4-1/2	4.2	144.7	3410	3.4	314	9.2	5320	3.4	329	6.2	6630	2.5	318	4.8

1. Class A: f'c = 5500 psi @ 90 days
   Class B: f'c = 4000 psi @ 28 days
   Class C: f'c = 3000 psi @ 90 days
   Class E: f'c = 2000 psi @ 28 days

2. Nonsafety-related only.

3. Correlation testing for 90-day compressive strength projections.

TABLE 3.8.1-7A

LOADING COMBINATION FOR DESIGN AND FINAL ANALYSIS
OF CONTAINMENT SHELL

	Loading Condition No.	STPEGS Project Criteria Category	Loading Combinations	Final Analysis Performed	Remarks
Service	1	Construction	$D + L + F_i + T_o$	Yes	Initial prestress case is more critical
Service	2	Tests	$D + L + F_i + P_t + T_t$	Yes	T_t is considered same as T_oI initial prestress is more critical
Service	3	Normal	$D + L + F + T_o + R_o + P_v$	Yes	R_o is a local load
Service	4	Severe Environment	$D + L + F + T_o + E_o + R_o + P_v$	Yes	R_o is a local load
Service	5	Severe Environment	$D + L + F + T_o + W + R_o + P_v$	No	Less severe than loading combination #4
Non-Service	6	Severe Environment	$D + 1.3L + F + T_o + 1.5E_o + R_o + P_v$	Yes	R_o is a local load
Non-Service	7	Severe Environment	$D + 1.3L + F + T_o + 1.5W + R_o + P_v$	No	Less severe than loading combination #6
Non-Service	8	Severe Environment	$D + L + F + T_o + E$ (or w) $+ R_o$	No	Less severe than loading combination #6
Non-Service	9	Extreme Environment	$D + L + F + T_o + E_{ss} + R_o + P_v$	Yes	R_o is a local load
Non-Service	10	Extreme Environment	$D + L + F + T_o + W_t + R_o + P_v$	Yes	R_o is a local load
Non-Service	11	Extreme Environment	$D + L + F + T_o + H + R_o + P_v$	Yes	R_o is a local load
Non-Service	12	Abnormal	$D + L + F + T_o + R_o + T_a + R_a + 1.5P_a$	Yes	R_o and R_a are local loads
Non-Service	13	Abnormal	$D + L + F + T_o + 1.25\ R_o + T_a + 1.25\ R_a + P_a$	No	Less critical than loading combination #12
Non-Service	14	Abnormal with Severe Environment	$D + L + F + T_o + R_o + T_a + R_a +$ $1.25\ P_a + 1.25\ E_o$	Yes	R_o and R_a are local loads
Non-Service	15	Abnormal with Severe Environment	$D + L + F + T_o + R_o + T_a + R_a +$ $1.25\ P_a + 1.25\ _w$	No	Less severe than loading combination #14
Non-Service	16	Abnormal with Extreme Environment	$D + L + F + T_o + R_o + T_a + R_a +$ $P_a + E_{ss} + Y$	Yes	R_o, R_a and Y are local loads

TABLE 3.8.1-7A (Continued)

LOADING COMBINATION FOR DESIGN AND FINAL ANALYSIS
OF CONTAINMENT SHELL

Notation

D	=	Dead load
L	=	Live load
F_I	=	Initial prestress
F	=	Final prestress
T_o	=	Normal operating temperature
P_a	=	Design Basis Accident Pressure Load
W	=	Wind load
R_o	=	Pipe reactions during normal pressure and tornado missiles) operating or shutdown conditions
R_a	=	Pipe reactions above normal operating loads (R_o)
T_a	=	Design Basis accident Thermal Load
P_t	=	Test pressure (= 1.15 P_a)
T_t	=	Test temperature (assumed equal to T_o)
P_v	=	Design external pressure (vacuum)
E_o	=	Operating basis earthquake
E_{ss}	=	Safe shutdown earthquake
W_t	=	Tornado loads (including differential
Y	=	Pipe rupture load
H	=	Flood load

Local loads are not considered in the overall analysis but are taken into account in local design.

Table 3.8.1-7B[a][b]

STRESS ANALYSIS RESULTS

$D + L + F_i + T_0$ (See Notations)
(Service Load)

CONSTRUCTION

Portion	Section	CONCRETE STRESSES Meridional Primary Mem	Meridional Primary Mem & Ben (PSI)	Meridional Primary and Secondary Mem (PSI)	Meridional Primary and Secondary Mem & Ben (PSI)	Hoop Primary Mem (PSI)	Hoop Primary Mem & Ben (PSI)	Hoop Primary and Secondary Mem (PSI)	Hoop Primary and Secondary Mem & Ben (PSI)	REINFORCEMENT STRESSES Meridional Primary In-side (KSI)	Meridional Primary Out-side (KSI)	Meridional Primary and Secondary In-side (KSI)	Meridional Primary and Secondary Out-side (KSI)	Hoop Primary In-side (KSI)	Hoop Primary Out-side (KSI)	Hoop Primary and Secondary In-side (KSI)	Hoop Primary and Secondary Out-side (KSI)	LINER[g] Strain Merid-ional $x10^{-6}$ in/in (KSI)	LINER[g] Strain Hoop $x10^{-6}$ in/in
Allow-able	Shell	-1925	-2475	-2475	-3300	-1925	-2475	-2475	-3300	±30	±30	±40	±40	±30	±30	±40	±40	±4000	±4000
	Basemat	-1400	-1800	-1800	-2400	-1400	-1800	-1800	-2400										
Dome	1	-1223	-1316	-1224	-1935	-1189	-1285	-1189	-1896	-8.2	-8.9	-8.7	-11.1	-7.9	-8.6	-8.4	-10.9	-834	-824
	2	-1725	-1890	-1725	-2542	-1446	-1627	-1451	-2747	-11.2	-12.9	-13.1	-16.0	-9.2	-11.1	-10.8	-14.5	-922	-860
	3	-1682	-1973	-1683	-3240	-1821	-2349	-1837	-3221	-12.8	-10.1	-17.0	-14.6	-10.6	-15.7	-14.8	-20.5	-1177	1049
Wall	4	-1262	-1374	-1262	-1915	-1601	-1836	-1638	-2411	-8.2	-9.4	-10.9	-12.4	-10.1	-12.6	-12.9	-15.9	-851	-923
	5	-1221	-1218	-1222	-2101	-1562	-1583	-1579	-2370	-8.5	-8.4	-11.6	-11.7	-10.6	-11	-13.3	-14.6	-899	-964
	6	-1130	-1128	-1113	-1720	-1694	-1695	-1741	-2333	-7.9	-7.7	-9.7	-10.3	-11.8	-11.5	-13.1	-15.4	-807	-955
	7	-1129	-2912(c)	-1151	-3456(e)	-391	-933	-413	-1000	-13.9	2.3	-14.7	9.1	-4.1	-.37	-3.0	3.3	-1229	-633
Basemat Slab	8	-33	-788	-46	-656	-24	-121	-36	-101	-5.4	22.2	-4.4	23.8	-.88	1.0	-.65	2.5	-561	-418
	9	-34	-71	-64	-144	(d)	-58	-10	-11	-.52	0	-.95	1.6	-.23	3.3	3.3	4.2	-430	-280
	10	-31	-48	-79	-131	-18	-18	-62	-58	-.36	-.11	-.9	.94	-.15	-.12	.5	-.42	-426	-395
	11	-29	-62	-121	-167	-12	-35	-93	-50	-.47	.03	-1.2	1.07	-.26	.15	-.33	.7	-436	-404

Footnotes:

(a) Sign Conventions are:
Stresses and Strains.......(+) tensile.......(-) compressive

(b) The stresses were obtained from OPTCON computer output.

(c) Actual cylinder breaks of concrete result in an allowable of 3060 psi.

(d) The section is assumed cracked when concrete stress is in tension.

(e) The concrete allowable stress is 4080 psi based on concrete test strength f_i = 6800 psi.

(f) The allowable bar stress can be increased 33 1/3 percent during test condition.

(g) Allowable liner strains shown are based on the lowest values from the lowest values from the ASME Code, Section III, Division 2.

Table 3.8.1-7B(a)(b) (Continued)

STRESS ANALYSIS RESULTS

D + L + F_i + Pt T_t (See Notations)
(Service Load)

TESTS

		CONCRETE STRESSES								REINFORCEMENT STRESSES								LINER(g)	
		Meridional				Hoop				Meridional				Hoop				Strain	
Portion	Section	Primary		Primary and Secondary		Primary		Primary and Secondary		Primary		Primary and Secondary		Primary		Primary and Secondary		Merid-ional $x10^{-6}$ in/in	Hoop $x10^{-6}$ in/in
		Mem (PSI)	Mem & Ben (PSI)	Mem (PSI)	Mem & Ben (PSI)	Mem (PSI)	Mem & Ben (PSI)	Mem (PSI)	Mem & Ben (KSI)	In-side (KSI)	Out-side (KSI)	In-side (KSI)	Out-side (KSI)	In-side (KSI)	Out-side (KSI)	In-side (KSI)	Out-side (KSI)		
Allow-able	Shell	-1925	-2475	-2475	-3300	-1925	-2475	-2475	-3300	±30	±30	±40	±40	±30	±30	±40	±40	±4000	±4000
	Basemat	-1400	-1800	-1800	-2400	-1400	-1800	-1800	-2400										
	1	-432	-531	-432	-1005	-403	-508	-403	-945	-2.7	-3.4	-2.3	3.6	-2.5	-3.2	-2.0	3.8	-635	-621
Dome	2	-963	-1145	-962	-1569	-665	-876	-671	-1258	-5.9	-7.6	-8.4	-6.6	-3.8	-5.8	-5.7	-4.7	-771	-696
	3	-923	-1022	-920	-2225	-747	-1368	-764	-1567	-6.8	-5.8	-10	5.8	-3.0	-8.7	-6.8	-6.2	-931	-771
	4	-68	-784	-678	-1383	-567	-830	-605	-1214	-4.2	-5.3	-7.0	-4.2	-2.9	-5.5	-5.7	-4.2	-726	-684
	5	-643	-648	-633	-1490	-361	-423	-378	-1010	-4.4	-4.5	-7.0	4.1	-2.3	-2.9	-3.6	6.5	-752	-636
Wall	6	-537	-565	-521	-1182	-439	-446	-497	-1242	-3.9	-3.5	-5.6	3.2	-3.1	-2.9	-3.4	11.8	-677	-693
	7	-597	-1344	-619	-894	-204	-357	-224	-283	-.78	-7.8	-4.9	-2.0	-1.0	-2.0	-1.2	-.11	-606	-459
	8	(d)	(d)	(d)	(d)	(d)	-378	(d)	-285	18.5	8.6	13.2	11.3	-2.3	13.5	-1.5	15.7	67.2	-469
Basemat	9	(d)	-1109	(d)	-1021	(d)	-1000	(d)	-874	-5.8	33.(f)	-5.	34.5	-5.3	35.(f)	4.3	36.1	-673	-633
Slab	10	(d)	-991	-20	-963	(d)	-985	-10	-942	-5.5	25.3	-5.2	26.7	-5.8	26.1	-5.4	27.4	-657	-651
	11	(d)	-953	-48	-962	(d)	-943	-24	-941	-5.1	26.1	-5.1	27.1	-5.4	27.8	-5.3	28.9	-656	-651

Footnotes:

(a) Sign Conventions are:
Stresses and Strains.......(+) tensile.......(-) compressive

(b) The stresses were obtained from OPTCON computer output.

(c) Actual cylinder breaks of concrete result in an allowable of 3060 psi.

(d) The section is assumed cracked when concrete stress is in tension.

(e) The concrete allowable stress is 4080 psi based on concrete test strength f_i = 6800 psi.

(f) The allowable bar stress can be increased 33 1/3 percent during test condition.

(g) Allowable liner strains shown are based on the lowest values from the lowest values from the ASME Code, Section III, Division 2.

Table 3.8.1-7B[(a)(b)] (Continued)

STRESS ANALYSIS RESULTS

D + L + F + T_0 + P_v (See Notations)
(Service Load)

Normal

Portion	Section	Concrete Stresses Meridional Primary Mem (PSI)	Concrete Stresses Meridional Primary Mem & Ben (PSI)	Concrete Stresses Meridional Primary and Secondary Mem (PSI)	Concrete Stresses Meridional Primary and Secondary Mem & Ben (PSI)	Concrete Stresses Hoop Primary Mem (PSI)	Concrete Stresses Hoop Primary Mem & Ben (PSI)	Concrete Stresses Hoop Primary and Secondary Mem (PSI)	Concrete Stresses Hoop Primary and Secondary Mem & Ben (PSI)	Reinforcement Stresses Meridional Primary In-side (KSI)	Reinforcement Stresses Meridional Primary Out-side (KSI)	Reinforcement Stresses Meridional Primary and Secondary In-side (KSI)	Reinforcement Stresses Meridional Primary and Secondary Out-side (KSI)	Reinforcement Stresses Hoop Primary In-side (KSI)	Reinforcement Stresses Hoop Primary Out-side (KSI)	Reinforcement Stresses Hoop Primary and Secondary In-side (KSI)	Reinforcement Stresses Hoop Primary and Secondary Out-side (KSI)	LINER[(g)] Strain Merid-ional $x10^{-6}$ in/in	LINER[(g)] Strain Hoop $x10^{-6}$ in/in
Allow-able	Shell	-1650	-2475	-2475	-3300	-1650	-2475	-2475	-3300										
										±30	±30	±40	±40	±30	±30	±40	±40	±4000	±4000
	Basemat	-1200	-1800	-1800	-2400	-1200	-1800	-1800	-2400										
Dome	1	-1106	-1185	-1107	-1793	-1073	-1155	-1073	-1755	-7.4	-8.0	-8.0	-10.3	-7.2	-7.8	-7.7	-10.0	-816	-806
	2	-1251	-1250	-1251	-1968	-1124	-1179	-1129	-1902	-8.7	-8.6	-10.8	-11.7	-8.0	-7.3	-9.5	-10.7	-868	-852
	3	-1195	-1363	-1195	-2627	-1432	-1621	-1449	-2964	-9.0	-7.3	-12.8	-11.6	-10.6	-8.6	-14.0	-13.7	-1029	-1110
Wall	4	-939	-1142	-939	-1649	-1277	-1303	-1315	-2113	-5.6	-7.7	-8.6	-10.5	-8.6	-9.1	-11.5	-12.4	-768	-902
	5	-1106	-1105	-1106	-1998	-1316	-1339	-1383	-2161	-7.7	-7.6	-10.8	-10.8	-8.9	-9.3	-11.7	-12.8	-874	-913
	6	-1017	-1029	-1018	-1642	-1445	-1453	-1492	-2122	-7.1	-6.9	-9.1	-9.3	-10.1	-9.8	-11.5	-13.6	-788	-904
	7	-1028	-2799(c)	-1050	-3297	-365	-894	-387	-947	-13.1	3.1	-13.7	9.9	-3.8	-.18	-2.8	3.4	-1191	-620
Basemat Slab	8	-35	-745	-47	-618	-18	-77	-29	-13	-5.1	20.4	-4.2	22.	-.56	.57	-.07	1.67	551	-391
	9	-35	-41	-64	-91	-13	-14	-34	-42	-.21	-.32	-.49	-.67	-.11	-.08	1.8	-.31	-409	-324
	10	-34	-55	-82	-99	-19	-76	-63	-102	-.1	-.42	-.58	-.72	.66	-.55	1.4	-.71	-411	-372
	11	-31	-33	-123	-113	-14	-33	-95	-60	-.2	-.26	-.83	-.62	.1	-.25	.68	-.44	-421	-389

Footnotes:

(a) Sign Conventions are:
Stresses and Strains.......(+) tensile.......(-) compressive
(b) The stresses were obtained from OPTCON computer output.
(c) Actual cylinder breaks of concrete result in an allowable of 3060 psi.
(d) The section is assumed cracked when concrete stress is in tension.
(e) The concrete allowable stress is 4080 psi based on concrete test strength f_i = 6800 psi.
(f) The allowable bar stress can be increased 33 1/3 percent during test condition.
(g) Allowable liner strains shown are based on the lowest values from the lowest values from the ASME Code, Section III, Division 2.

Table 3.8.1-7B[(a)(b)] (Continued)

STRESS ANALYSIS RESULTS

D + L + F + T_0+ E_0+ P_v (See Notations)
(Service Load)

SEVERE ENVIRONMENT

		CONCRETE STRESSES								REINFORCEMENT STRESSES								$LINER_{(g)}$ Strain	
		Meridional				Hoop				Meridional				Hoop					
Portion	Section	Primary		Primary and Secondary		Primary		Primary and Secondary		Primary		Primary and Secondary		Primary		Primary and Secondary		Merid-ional $x10^{-6}$ in/in	Hoop $x10^{-6}$ in/in
		Mem (PSI)	Mem & Ben (PSI)	Mem (PSI)	Mem & Ben (PSI)	Mem (PSI)	Mem & Ben (PSI)	Mem (PSI)	Mem & Ben (PSI)	In-side (KSI)	Out-side (KSI)	In-side (KSI)	Out-side (KSI)	In-side (KSI)	Out-side (KSI)	In-side (KSI)	Out-side (KSI)		
Allow-able	Shell	-1650	-2475	-2475	-3300	-1650	-2475	-2475	-3300										
										±30	±30	±40	±40	±30	±30	±40	±40	±4000	±4000
	Basemat	-1200	-1800	-1800	-2400	-1200	-1800	-1800	-2400										
Dome	1	-1110	-1189	-1110	-1797	-1076	-1159	-1076	-1759	-7.4	-8.0	-7.7	-10.3	-7.2	-7.8	-7.4	-10.0	-807	-796
	2	-1258	-1257	-1258	-1926	-1133	-1188	-1132	-1849	-8.8	-8.6	-10.5	-11.7	-8.0	-7.4	-9.1	-10.8	-851	-838
	3	-1236	-1414	-1231	-2679	-1473	-1720	-1490	-3049	-9.3	-7.6	-13.1	-11.1	-11.0	-8.6	-14.5	-13.3	-1041	-1131
Wall	4	-966	-1183	-966	-1691	-1310	-1351	-1298	-2045	-5.7	-7.9	-8.3	-10.7	-8.8	-9.4	-10.9	-12.7	-759	-886
	5	-1155	-1153	-1145	-2044	-1351	-1370	-1368	-2194	-8.0	-7.9	-11.1	-10.2	-9.2	-9.6	-12.	-12.1	-886	-921
	6	-1088	-1099	-1089	-1557	-1490	-1497	-1538	-2078	-7.6	-7.4	-8.5	-9.9	-10.4	-10.1	-10.8	-13.9	-767	-881
	7	-1124	-2991(c)	-1146	-3505(e)	-439	-1000	-459	-1092	-14.1	2.9	-14.8	9.7	-4.5	-.70	-3.4	2.9	-1241	-655
Basemat Slab	8	-50	-796	-62	-635	-36	-89	-35	-64	-5.5	20.4	-4.2	22.1	-58	2.6	1.4	4.1	-566	-345
	9	-52	-71	-52	-124	-19	-101	-25	-67	-.47	.65	.23	1.3	-.62	7.2	.59	2.9	-400	-375
	10	-45	-97	-68	-137	-28	-133	-50	-152	-.49	-.72	-.76	1.5	1.4	-.95	1.9	1.4	-423	-378
	11	-32	-62	-118	-157	-15	-52	-90	-75	-.45	-.33	-1.1	1.1	.35	.38	.57	1.3	-433	-400

Footnotes:
(a) Sign Conventions are:
Stresses and Strains.......(+) tensile.......(-) compressive
(b) The stresses were obtained from OPTCON computer output.
(c) Actual cylinder breaks of concrete result in an allowable of 3060 psi.
(d) The section is assumed cracked when concrete stress is in tension.
(e) The concrete allowable stress is 4080 psi based on concrete test strength f_i = 6800 psi.
(f) The allowable bar stress can be increased 33 1/3 percent during test condition.
(g) Allowable liner strains shown are based on the lowest values from the lowest values from the ASME Code, Section III, Division 2.

Table 3.8.1-7B[(a)(b)] (Continued)

STRESS ANALYSIS RESULTS

D + 1.3L + F + T_0+ 1.5E_0+ P_v (See Notations)
(Non-Service Load)

SEVREE ENVIRONMENT

		CONCRETE STRESSES								REINFORCEMENT STRESSES								LINER(g)	
		Meridional				Hoop				Meridional				Hoop				Strain	
Portion	Section	Primary		Primary and Secondary		Primary		Primary and Secondary		Primary		Primary and Secondary		Primary		Primary and Secondary		Merid-ional $x10^{-6}$ in/in	Hoop $x10^{-6}$ in/in
		Mem (PSI)	Mem & Ben (PSI)	Mem (PSI)	Mem & Ben (PSI)	Mem (PSI)	Mem & Ben (PSI)	Mem (PSI)	Mem & Ben (PSI)	In-side (KSI)	Out-side (KSI)	In-side (KSI)	Out-side (KSI)	In-side (KSI)	Out-side (KSI)	In-side (KSI)	Out-side (KSI)		
Allow-	Shell	-3300	-4125	-4125	-4675	-3300	-4125	-4125	-4675										
able										±54	±54	±54	±54	±54	±54	±54	±54	±10000	±10000
	Basemat	-2400	-3000	-3000	-3400	-2400	-3000	-3000	-3400										
	1	-1115	-1196	-1115	-1766	-1079	-1164	-1079	-1729	-7.1	-7.7	-7.4	-10.0	-6.8	-7.4	-7.1	-9.7	-801	-790
Dome	2	-1263	-1262	-1263	-1887	-1141	-1196	-1140	-1835	-8.4	-8.3	-10.2	-11.4	-7.7	-7.0	-8.9	-10.4	-848	-834
	3	-1253	-1424	-1248	-2543	-1490	-1758	-1506	-2848	-9.0	-7.3	-13.2	-10.6	-11.0	-8.1	-14.8	-13.0	-1044	-1144
	4	-979	-1214	-979	-1692	-1327	-1369	-1314	-1992	-5.3	-7.7	-7.9	-10.6	-8.5	-9.2	-10.6	-12.5	-748	-877
	5	-1188	-1188	-1178	-2017	-1364	-1392	-1380	-2121	-7.9	-7.7	-11.1	-9.7	-8.9	-9.3	-11.8	-11.6	-884	-914
Wall	6	-1107	-1139	-1108	-1538	-1536	-1555	-1583	-2064	-7.4	-7.0	-7.9	-9.5	-10.5	-10.0	-10.5	-14.0	-753	-870
	7	-1183	-3133	-1204	-3394	-507	-1138	-527	-1241	-16.1	5.5	-16.5	11.4	-4.8	1.0	-3.8	2.8	-1349	-678
	8	-66	-834	-78	-676	-51	-91	-49	-64	-6.6	22.8	-5.1	22.5	-.75	3.6	1.8	5.1	-601	-334
Basemat	9	-71	-107	-70	-132	-27	-127	-27	-73	-.72	1.7	-.47	2.8	-.82	3.4	.51	4.0	-415	-398
Slab	10	-61	-114	-72	-155	-39	-143	-51	-162	-.77	.92	-1.0	2.2	1.3	1.4	1.9	2.4	-434	-396
	11	-44	-76	-124	-176	-22	-55	-92	-74	-.59	-.4	-1.3	1.1	-.4	.67	.49	1.3	-443	-409

Footnotes:
(a) Sign Conventions are:
Stresses and Strains.......(+) tensile.......(-) compressive
(b) The stresses were obtained from OPTCON computer output.
(c) Actual cylinder breaks of concrete result in an allowable of 3060 psi.
(d) The section is assumed cracked when concrete stress is in tension.
(e) The concrete allowable stress is 4080 psi based on concrete test strength f_i = 6800 psi.
(f) The allowable bar stress can be increased 33 1/3 percent during test condition.
(g) Allowable liner strains shown are based on the lowest values from the lowest values from the ASME Code, Section III, Division 2.

Table 3.8.1-7B[(a)(b)] (Continued)

STRESS ANALYSIS RESULTS

D + L + F + T_0+ E_{ss} (orW_t)+ P_v (See Notations)
(Non-Service Load)

EXTREME ENVIRONMENT

Portion	Section	CONCRETE STRESSES Meridional Primary Mem (PSI)	Meridional Primary Mem & Ben (PSI)	Meridional Primary and Secondary Mem (PSI)	Meridional Primary and Secondary Mem & Ben (PSI)	Hoop Primary Mem (PSI)	Hoop Primary Mem & Ben (PSI)	Hoop Primary and Secondary Mem (PSI)	Hoop Primary and Secondary Mem & Ben (PSI)	REINFORCEMENT STRESSES Meridional Primary In-side (KSI)	Meridional Primary Out-side (KSI)	Meridional Primary and Secondary In-side (KSI)	Meridional Primary and Secondary Out-side (KSI)	Hoop Primary In-side (KSI)	Hoop Primary Out-side (KSI)	Hoop Primary and Secondary In-side (KSI)	Hoop Primary and Secondary Out-side (KSI)	LINER(g) Strain Merid-ional x10^-6 in/in	LINER(g) Strain Hoop x10^-6 in/in
Allow-able	Shell	-3300	-4125	-4125	-4675	-3300	-4125	-4125	-4675	±54	±54	±54	±54	±54	±54	±54	±54	±10000	±10000
	Basemat	-2400	-3000	-3000	-3400	-2400	-3000	-3000	-3400										
Dome	1	-1207	-1300	-1207	-1871	-1158	-1268	-1158	-1831	-7.7	-8.4	-7.4	-10.7	-7.3	-8.1	-7.1	-10.4	-800	-790
	2	-1268	-1290	-1292	-1886	-1183	-1222	-1181	-1829	-8.6	-8.5	-10.2	-11.7	-7.9	-7.4	-8.9	-10.8	-848	-833
	3	-1347	-1484	-1342	-2601	-1529	-1815	-1546	-2887	-9.6	-8.1	-13.9	-10.5	-11.3	-8.6	-15.	-13.	-1062	-1158
Wall	4	-1039	-1263	-1040	-1743	-1347	-1407	-1334	-1982	-5.8	-8.1	-7.9	-11.0	-8.6	-9.4	10.5	-12.8	-750	-874
	5	-1271	-1290	-1260	-2049	-1435	-1551	-1452	-2146	-8.2	-8.6	-11.5	-9.5	-9.1	-10.4	-12.	-11.5	-893	-921
	6	-1162	-1170	-1162	-1599	-1553	-1573	-1600	-2152	-7.7	-7.7	-7.9	-10.2	-10.4	-10.7	-10.4	-14.7	-754	-874
	7	-1229	-2993	-1250	-3313	-395	-1103	-538	-1242	-15.7	3.5	-16.4	9.4	-4.8	1.3	-3.9	3.4	-1316	-678
Basemat Slab	8	-69	-790	-81	-640	-60	-102	-58	-64	-6.3	21.4	-4.9	22.7	-.82	4.3	2.9	5.8	-589	-323
	9	-73	-101	-72	-151	-30	-142	-30	-107	-.68	2.9	1.3	4.1	1.3	3.9	1.6	4.6	-426	-414
	10	-60	-143	-81	-177	-42	-183	-58	-195	-.78	1.9	-.9	3.4	2.6	2.2	2.7	3.3	-433	-402
	11	-38	-81	-119	-166	-21	-79	-91	-96	-.62	-.44	-1.2	1.3	1.	.96	1.	1.6	-439	-411

Footnotes:
(a) Sign Conventions are:
Stresses and Strains.......(+) tensile.......(-) compressive
(b) The stresses were obtained from OPTCON computer output.
(c) Actual cylinder breaks of concrete result in an allowable of 3060 psi.
(d) The section is assumed cracked when concrete stress is in tension.
(e) The concrete allowable stress is 4080 psi based on concrete test strength f_i = 6800 psi.
(f) The allowable bar stress can be increased 33 1/3 percent during test condition.
(g) Allowable liner strains shown are based on the lowest values from the lowest values from the ASME Code, Section III, Division 2.

Table 3.8.1-7B[(a)(b)] (Continued)

STRESS ANALYSIS RESULTS

$D + L + F + T_0 + H + P_v$ (See Notations)
(Non-Service Load)

EXTREME ENVIRONMENT

Portion	Section	CONCRETE STRESSES Meridional Primary Mem (PSI)	Mem & Ben (PSI)	Meridional Primary and Secondary Mem (PSI)	Mem & Ben (PSI)	Hoop Primary Mem (PSI)	Mem & Ben (PSI)	Hoop Primary and Secondary Mem (PSI)	Mem & Ben (PSI)	REINFORCEMENT STRESSES Meridional Primary In-side (KSI)	Out-side (KSI)	Meridional Primary and Secondary In-side (KSI)	Out-side (KSI)	Hoop Primary In-side (KSI)	Out-side (KSI)	Hoop Primary and Secondary In-side (KSI)	Out-side (KSI)	LINER[(g)] Strain Merid-ional $x10^{-6}$ in/in	Hoop $x10^{-6}$ in/in
Allow-able	Shell	-3300	-4125	-4125	-4675	-3300	-4125	-4125	-4675	±54	±54	±54	±54	±54	±54	±54	±54	±10000	±10000
	Basemat	-2400	-3000	-3000	-3400	-2400	-3000	-3000	-3400										
Dome	1	-1113	-1192	-1114	-1756	-1079	-1162	-1078	-1721	-7.0	-7.6	-7.7	-9.9	-6.8	-7.4	-7.5	-9.6	-810	-801
	2	-1261	-1259	-1260	-1933	-1133	-1189	-1138	-1878	-8.4	-8.2	-10.6	-11.3	-7.6	-6.9	-9.3	-10.3	-861	-846
	3	-1214	-1388	-1209	-2514	-1444	-1631	-1461	-2765	-8.8	-7.0	-12.9	-11.2	-10.4	-8.3	-14.2	-13.4	-1034	-1116
Wall	4	-961	-1164	-961	-1646	-1287	-1317	-1324	-2062	-5.3	-7.4	-8.2	-10.2	-8.3	-8.8	-11.3	-12.1	-759	-897
	5	-1159	-1159	-1159	-1947	-1337	-1382	-1354	-2139	-7.7	-7.5	-10.4	-10.8	-8.9	-9.2	-11.8	-12.7	-864	-919
	6	-1090	-1101	-1090	-1652	-1536	-1562	-1584	-2137	-7.0	-7.2	-8.8	-9.7	-10.3	-10.6	-11.6	-14.5	-783	-914
	7	-1057	-3403	-1079	-3687	-381	-1204	-403	-994	-14.1	9.1	-14.2	15.5	-3.6	3.2	-2.6	5.0	-1296	-617
Basemat Slab	8	-59	-793	-71	-637	-37	-55	-35	-70	-6.3	21.	-4.8	22.2	-.45	.82	2.0	2.3	-588	-322
	9	-56	-99	-80	-142	-11	-93	-30	-86	-.32	-.80	-.94	-1.1	4.5	-.55	4.1	.56	-426	-238
	10	-49	-108	-94	-144	-20	-173	-62	-171	2.7	-.86	.68	-1.1	4.4	-1.2	3.8	-1.2	-407	-288
	11	-46	-83	-134	-624	-17	-124	-95	-130	0.1	-.67	-1.0	-.96	3.0	-.88	2.3	-.96	-427	-352

Footnotes:
(a) Sign Conventions are:
Stresses and Strains.......(+) tensile.......(-) compressive
(b) The stresses were obtained from OPTCON computer output.
(c) Actual cylinder breaks of concrete result in an allowable of 3060 psi.
(d) The section is assumed cracked when concrete stress is in tension.
(e) The concrete allowable stress is 4080 psi based on concrete test strength f_i = 6800 psi.
(f) The allowable bar stress can be increased 33 1/3 percent during test condition.
(g) Allowable liner strains shown are based on the lowest values from the lowest values from the ASME Code, Section III, Division 2.

Table 3.8.1-7B[a][b] (Continued)

STRESS ANALYSIS RESULTS

D + L + F + T_0+ T_a + 1.5P_a (See Notations)
(Non-Service Load)

ABNORMAL

Portion	Section	CONCRETE STRESSES Meridional Primary Mem (PSI)	Meridional Primary Mem & Ben (PSI)	Meridional Primary and Secondary Mem (PSI)	Meridional Primary and Secondary Mem & Ben (PSI)	Hoop Primary Mem (PSI)	Hoop Primary Mem & Ben (PSI)	Hoop Primary and Secondary Mem (PSI)	Hoop Primary and Secondary Mem & Ben (PSI)	REINFORCEMENT STRESSES Meridional Primary In-side (KSI)	Meridional Primary Out-side (KSI)	Meridional Primary and Secondary In-side (KSI)	Meridional Primary and Secondary Out-side (KSI)	Hoop Primary In-side (KSI)	Hoop Primary Out-side (KSI)	Hoop Primary and Secondary In-side (KSI)	Hoop Primary and Secondary Out-side (KSI)	LINER(g) Strain Merid-ional x10^{-6} in/in	LINER(g) Strain Hoop x10^{-6} in/in
Allow-able	Shell	-3300	-4125	-4125	-4675	-3300	-4125	-4125	-4675	±54	±54	±54	±54	±54	±54	±54	±54	±10000	±10000
	Basemat	-2400	-3000	-3000	-3400	-2400	-3000	-3000	-3400										
Dome	1	-42	-211	-43	(d)	-17	-341	-17	(d)	1.2	-.48	21.4	15.2	5.7	.52	22.8	16.2	-557	-497
	2	-223	-224	-223	-464	-71	-120	-79	(d)	-1.3	-1.4	12.3	.88	-.56	-.1	21.2	15.9	-1295	-662
	3	-172	-252	-169	-681	(d)	-265	(d)	-6	-.7	-1.4	.26	13.7	1.8	10.2	8.6	21.2	-545	-273
Wall	4	-156	-352	-154	-636	(d)	(d)	(d)	(d)	-.08	-1.9	16.8	-1.4	34.5	12.4	34.7	30.4	-1043	-225
	5	-327	-334	-317	-1028	(d)	(d)	(d)	(d)	-1.9	-2.1	-3.2	8.7	49.3	19.7	20.9	40.8	-626	-210
	6	-220	-273	-204	-533	(d)	(d)	(d)	(d)	-1.6	-1.	9.5	-1.7	28.5	19.6	35.3	36.7	-1288	-268
	7	-309	-2610	-331	-1451	-120	-1213	-140	-440	19.6	-12.2	5.6	-7.0	14.7	-.93	2.8	-1.1	-56	-217
Basemat Slab	8	(d)	-7	(d)	(d)	(d)	-371	(d)	-255	40.1	2.4	22.9	6.1	-2.3	17.8	-1.2	20.	419	-468
	9	(d)	-1257	d	-1151	d	-1180	(d)	-1052	-7.9	43.5	-6.7	45.1	-7.8	42.1	-6.5	43.2	-765	-730
	10	(d)	-1147	(d)	-1097	(d)	-1149	(d)	-1087	-7.5	33.4	-6.9	34.8	-8.	34.5	-7.3	35.7	-746	-742
	11	(d)	-1098	-25	-1081	(d)	-1096	(d)	-1070	-7.	34.3	-6.8	35.4	-7.4	36.3	-7.1	37.4	-740	-736

Footnotes:
(a) Sign Conventions are:
Stresses and Strains.......(+) tensile.......(-) compressive
(b) The stresses were obtained from OPTCON computer output.
(c) Actual cylinder breaks of concrete result in an allowable of 3060 psi.
(d) The section is assumed cracked when concrete stress is in tension.
(e) The concrete allowable stress is 4080 psi based on concrete test strength f_i = 6800 psi.
(f) The allowable bar stress can be increased 33 1/3 percent during test condition.
(g) Allowable liner strains shown are based on the lowest values from the lowest values from the ASME Code, Section III, Division 2.

Table 3.8.1-7B[(a)(b)] (Continued)

STRESS ANALYSIS RESULTS

$D + L + F + T_0 + E_0 + T_a + 1.25P_a$ (See Notations)
(Non-Service Load)

ABNORMAL/SEVERE ENVIRONMENT

Portion	Section	CONCRETE STRESSES Meridional Primary Mem (PSI)	Meridional Primary Mem & Ben (PSI)	Meridional Primary and Secondary Mem (PSI)	Meridional Primary and Secondary Mem & Ben (PSI)	Hoop Primary Mem (PSI)	Hoop Primary Mem & Ben (PSI)	Hoop Primary and Secondary Mem (PSI)	Hoop Primary and Secondary Mem & Ben (PSI)	REINFORCEMENT STRESSES Meridional Primary In-side (KSI)	Meridional Primary Out-side (KSI)	Meridional Primary and Secondary In-side (KSI)	Meridional Primary and Secondary Out-side (KSI)	Hoop Primary In-side (KSI)	Hoop Primary Out-side (KSI)	Hoop Primary and Secondary In-side (KSI)	Hoop Primary and Secondary Out-side (KSI)	LINER[(g)] Strain Merid-ional $x10^{-6}$ in/in	LINER[(g)] Strain Hoop $x10^{-6}$ in/in
Allow-able	Shell	-3300	-4125	-4125	-4675	-3300	-4125	-4125	-4675	$\pm$54	$\pm$54	$\pm$54	$\pm$54	$\pm$54	$\pm$54	$\pm$54	$\pm$54	$\pm$10000	$\pm$10000
	Basemat	-2400	-3000	-3000	-3400	-2400	-3000	-3000	-3400										
Dome	1	-216	-297	-216	-624	-190	-276	-189	-571	-1.1	-1.6	12.7	3.6	-.94	-1.5	13.7	7.3	-1051	-902
	2	-396	-397	-396	-912	-250	-301	-254	-481	-2.4	-2.5	6.4	-3.8	-1.7	-1.2	11.1	1.6	-1471	-1253
	3	-386	-410	-381	-1299	-264	-462	-280	-1084	-2.3	-2.5	-3.2	12.7	-2.2	-.67	-.38	18.2	-695	-624
Wall	4	-314	-522	-312	-1005	-147	-216	-158	-451	-1.1	-3.0	10.8	-4.3	-.6	-1.3	20.6	12.3	-1241	-705
	5	-511	-396	-501	-1329	(d)	-205	-17	(d)	-3.2	-3.3	-5.4	6.1	28.7	7.0	14.0	14.5	-700	-706
	6	-435	-481	-420	-494	-78	-84	-129	(d)	-2.9	-2.4	-2.4	-1.6	8.6	7.7	22.2	15.0	-1578	-614
	7	-547	-2056	-569	-959	-253	-1099	-273	-358	12.7	-9.5	-2.8	-4.7	18.3	-2.3	3.5	-.25	-485	-468
Basemat Slab	8	(d)	(d)	(d)	(d)	(d)	-371	(d)	-290	34.2	6.6	20.2	10.1	-2.5	17.	-1.7	18.7.	320	-478
	9	(d)	-1071	-9	-988	(d)	-1039	(d)	-921	-6.8	39.	-5.9	40.7	-6.8	37.2	-5.6	38.4	-707	-684
	10	(d)	-1012	-27	-972	(d)	-1006	-14	-953	-6.5	29.9	-6.0	31.4	-6.9	30.8	-6.3	32.1	-701	-694
	11	(d)	-948	-41	-944	(d)	-943	-18	-930	-6.0	28.9	-5.7	30.	-6.3	30.7	-6.1	32.	-692	-677

Footnotes:
(a) Sign Conventions are:
Stresses and Strains.......(+) tensile.......(-) compressive
(b) The stresses were obtained from OPTCON computer output.
(c) Actual cylinder breaks of concrete result in an allowable of 3060 psi.
(d) The section is assumed cracked when concrete stress is in tension.
(e) The concrete allowable stress is 4080 psi based on concrete test strength f_i = 6800 psi.
(f) The allowable bar stress can be increased 33 1/3 percent during test condition.
(g) Allowable liner strains shown are based on the lowest values from the lowest values from the ASME Code, Section III, Division 2.

Table 3.8.1-7B[(a)(b)] (Continued)

STRESS ANALYSIS RESULTS

$D + L + F + T_0 + E_{ss} + T_a + P_a$ (See Notations)
(Non-Service Load)

Portion	Section	CONCRETE STRESSES								REINFORCEMENT STRESSES								LINER[(g)] Strain	
		Meridional				Hoop				Meridional				Hoop					
		Primary		Primary and Secondary		Primary		Primary and Secondary		Primary		Primary and Secondary		Primary		Primary and Secondary		Merid-ional $x10^{-6}$ in/in	Hoop $x10^{-6}$ in/in
		Mem (PSI)	Mem & Ben (PSI)	Mem (PSI)	Mem & Ben (PSI)	Mem (PSI)	Mem & Ben (PSI)	Mem (PSI)	Mem & Ben (PSI)	In-side (KSI)	Out-side (KSI)	In-side (KSI)	Out-side (KSI)	In-side (KSI)	Out-side (KSI)	In-side (KSI)	Out-side (KSI)		
Allow-able	Shell	-3300	-4125	-4125	-4675	-3300	-4125	-4125	-4675	±54	±54	±54	±54	±54	±54	±54	±54	±10000	±10000
	Basemat	-2400	-3000	-3000	-3400	-2400	-3000	-3000	-3400										
Dome	1	-388	-471	-388	-1021	-360	-448	-359	-971	-2.2	-2.7	7.2	-2.0	-2.0	-2.6	8.3	-1.5	-1554	-1303
	2	-566	-567	-565	-878	-424	-477	-428	-920	-3.5	-3.6	-3.8	-3.3	-2.8	-2.3	5.1	-4.5	-1668	-1475
	3	-580	-591	-575	-1651	-524	-782	-541	-1698	-3.7	-3.6	-5.8	10.2	-4.2	-2.1	-4.0	14.	-786	-786
Wall	4	-459	-681	-457	-1207	-393	-471	-404	-1009	-2.0	-4.1	6.0	-6.2	-2.1	-2.9	5.4	-5.5	-1392	-1425
	5	-672	-672	-662	-1536	-284	-333	-300	-885	-4.2	-4.3	-7.1	4.2	-1.5	-2.0	-2.2	8.0	-753	-591
	6	-615	-655	-600	-929	-381	-390	-432	-725	-4.1	-3.6	-4.6	-.75	-2.4	-2.3	10.3	-4.1	-1680	-1357
	7	-737	-1369	-758	-1448	-349	-722	-369	-719	+4.1	-6.7	-6.9	-1.8	11.3	-2.3	3.1	1.0	-730	-551
Basemat Slab	8	(d)	(d)	(d)	(d)	(d)	-340	(d)	-284	25.9	10.1	16.4	13.4	-2.4	15.5	-1.8	17.1	179	-476
	9	-13	-881	-35	-815	(d)	-881	(d)	-774	-5.7	33.2	-5.0	35.1	-5.6	31.5	-4.6	32.8	-647	-634
	10	(d)	-860	-47	-831	(d)	-845	-32	-803	-5.4	25.5	-5.0	27.1	-5.6.	26.1	-5.2	27.5	-652	-643
	11	(d)	-791	-57	-800	(d)	-781	-33	-781	-4.9	23.3	-4.9	24.4	-5.1	25.	-5.	26.3	-642	-636

Footnotes:
(a) Sign Conventions are:
Stresses and Strains.......(+) tensile.......(-) compressive
(b) The stresses were obtained from OPTCON computer output.
(c) Actual cylinder breaks of concrete result in an allowable of 3060 psi.
(d) The section is assumed cracked when concrete stress is in tension.
(e) The concrete allowable stress is 4080 psi based on concrete test strength f_i = 6800 psi.
(f) The allowable bar stress can be increased 33 1/3 percent during test condition.
(g) Allowable liner strains shown are based on the lowest values from the lowest values from the ASME Code, Section III, Division 2.

TABLE 3.8.1-7B [a] [b] (Continued)

STRESS ANALYSIS RESULTS

Notations

D	=	Dead load	T_a	=	Design Basis Accident Thermal Load
L	=	Live load	P_t	=	Test pressure (=1.15 p_a)
F_I	=	Initial prestress	T_t	=	Test temperature (assumed equal to T_o)
F	=	Final prestress	P_v	=	Design external pressure (vacuum)
T_o	=	Normal operating temperature	E_o	=	Operating basis earthquake
P_a	=	Design Basis Accident Pressure Load	E_{ss}	=	Safe shutdown earthquake
W	=	Wind load	W_t	=	Tornado loads (including differential pressure and tornado missiles)
R_o	=	Pipe reactions during normal operating or shutdown conditions	Y	=	Pipe rupture load
R_a	=	Pipe reactions above normal operating loads (R_o)	H	=	Flood load

Note: Local loads are not considered in the overall analysis but are taken into account in local design.

(a) Sign Conventions are:
Stresses and Strains.......(+) tensile.......(-) compressive
(b) The stresses were obtained from OPTCON computer output.
(c) Actual cylinder breaks of concrete result in an allowable of 3060 psi.
(d) The section is assumed cracked when concrete stress is in tension.
(e) The concrete allowable stress is 4080 psi based on concrete test strength f_i = 6800 psi.
(f) The allowable bar stress can be increased 33 1/3 percent during test condition.
(g) Allowable liner strains shown are based on the lowest values from the lowest values from the ASME Code, Section III, Division 2.

TABLE 3.8.2-1

LOAD COMBINATIONS FOR CLASS MC COMPONENTS

LOADING COMBINATIONS	D	L	F	Pa	Pt	Ro	Ra	To	Ta	Eo	Ess	Y	Pv	Tt
A	1	1	1			1		1		1				
B	1	1	1	1		1					1			
C	1	1	1			1							1	
D	1	1	1	1		1	1	1	1	1				
E	1	1	1	1		1	1	1	1		1			
F	1	1	1	1		1	1	1	1		1	1		
G	1				1									1

TABLE 3.8.2-2

PERSONNEL AIRLOCK PENETRATIONS

Quantity	Description
	Airlock to Meab
4	Air Supply
1	Electrical
1	Emergency Air and Pressure Test Connection
1	Handwheel Shaft
1	View Port
1	Equalizing Valve
	Airlock to Containment
1	Manual Door Operator
1	Electrical
1	View Port
1	Handwheel Shaft
1	Seal Air Supply
1	Pressure Relief
1	Equalizing Valve

TABLE 3.8.3-1

LOAD COMBINATIONS FOR CONCRETE INTERNAL STRUCTURES[b,c]

LOADS

Category	Loading Condition	No.	D	L	T_o	R_o	E_o	E_{ss}	T_a	R_a	P_a	Y[f]	Strength (See Note a)
Service	Normal	1	1.4	1.7									U for All Combinations
		2[d]	1.4	1.7	1.7	1.7							
	Severe Environmental	3	1.4	1.7			1.9						
		4[d]	1.4	1.7	1.7	1.7	1.9						
		5	1.2				1.9						
Non-Service	Abnormal	6[e]	1.0	1.0	1.0	1.0			1.0	1.0	1.5		
	Abnormal/Severe Environmental	7[e]	1.0	1.0	1.0	1.0	1.25		1.0	1.0	1.25	1.0	
	Extreme Environmental	8	1.0	1.0	1.0	1.0		1.0					
	Abnormal/Extreme Environmental	9[e]	1.0	1.0	1.0	1.0		1.0	1.0	1.0	1.0	1.0	

a U is the section strength required to resist design loads and is based on methods described in ACI 318-71.

b Loads not applicable to a particular system under consideration may be deleted.

c. If the effect of a nonpermanent load reduces the effect of others in the combination, the case of it being absent shall also be considered.

d. The values of load factors in combination no. 2 and no. 4 are multiplied by a factor of 0.75 to account for T_o and R_o.

e. In combinations 6, 7 and 9 the maximum values of P_a, $(T_o + T_a)$, $(R_o + R_a)$ and Y including an appropriate dynamic load factor shall be used unless a time history analysis is performed to justify otherwise.

f. In determining appropriate loads for Y, elasto-plastic behavior may be assumed with a maximum ductility ratio as stated in Table 3.5-13, provided excessive deflections do not result in loss of function of any safety-related system.

TABLE 3.8.3-2

LOAD COMBINATIONS FOR STEEL INTERNAL STRUCTURES [b,c]

LOADS

Category	Loading Condition	No.	D	L	T_o	R_o	E_o	E_{ss}	T_a	R_a	P_a	Y[f]	Strength (See Notes)
Service	Normal	1	1.0	1.0									S^c
		2	1.0	1.0	1.0	1.0							$1.33S^{(c)}$
	Severe Environmental	3	1.0	1.0			1.0						S^c
		4	1.0	1.0	1.0	1.0	1.0						$1.33S^{(c)}$
Non-Service	Abnormal	5[e]	1.0	1.0	1.0	1.0			1.0	1.0	1.0		$1.60S^{(c)}$
	Abnormal/Severe Environmental	6[e]	1.0	1.0	1.0	1.0	1.0		1.0	1.0	1.0	1.0	$1.60S^{(d)}$
	Extreme Environmental	7	1.0	1.0	1.0	1.0		1.0					$1.60S^{(c)}$
	Abnormal/Extreme Environmental	8[e]	1.0	1.0	1.0	1.0		1.0	1.0	1.0	1.0	1.0	$1.70S^{(d)}$

a. Loads not applicable to a particular system under consideration may be deleted.

b. If the effect of a nonpermanent load reduces the effect of others in combination, the case of it being absent shall also be considered.

c. "S" is the required strength based on the elastic design methods and allowable stresses defined in Part I of the AISC Specifications.

d. For these two combinations, in computing the required strength "S", the plastic section modulus of steel shapes may be used.

e. In combinations 5, 6, and 8, the maximum values of P_a, $(T_o + T_a)$, $(R_o + R_a)$ and Y including an appropriate dynamic load factor shall be used unless a time history analysis is performed to justify otherwise.

f. In determining appropriate loads for Y, elasto-plastic behavior may be assumed with a maximum ductility ratio as stated in Table 3.5-13, provided excessive deflections do not result in loss of function of any safety-related system.

TABLE 3.8.3-3

LOAD COMBINATIONS FOR CLASS 1, 2, 3, AND MC COMPONENT SUPPORTS

Loading Condition	No.	D	L	T_o	R_o	E_o	T_a	R_a	P_a	Y	E_{ss}
Normal Operating	1	1.0	1.0	1.0	1.0						
Upset	2	1.0	1.0	1.0	1.0	1.0					
Faulted	3	1.0	1.0	1.0	1.0		1.0	1.0	1.0	1.0	1.0

TABLE 3.8.3-4

CONTAINMENT INTERNAL STRUCTURES SUMMARY OF GOVERNING COMBINED STRESS RATIOS FROM THE BEAM/COLUMN INTERACTION EQUATION FOR PRINCIPAL STRUCTURAL STEEL MEMBERS

Description of Principal Members	Location of Principal Members	Governing Load Combination Number	Combined Stress Ratio (<1.0)
W27x160 BEAM	El. 68'-0" Between AZ 180° and AZ 206° 30'	1 [(a)]	0.95
W33x240 BEAM	El. 68'-0" at AZ 355°	8 [(a)]	0.54
W33x240 BEAM	El. 68'-0" at AZ 322.5°	1 [(a)]	0.58
W27x160 BEAM	El. 68'-0" Between AZ 5° and AZ 25.5°	1 [(a)]	0.68
W27x160 BEAM	El. 68'-0" Between AZ 334.5° and AZ 355°	1 [(a)]	0.72
W30x190 BEAM	El. 68'-0" Between AZ 106.5° and AZ 159.5°	5 [(a)]	0.97
W24x100 BEAM	El. 68'-0" Between AZ 127° and AZ 139°	1 [(a)]	0.87
W33x240 BEAM	El. 68'-0" at AZ 139°	5 [(a)]	0.86
W24x120 BEAM	El. 37'-3" at AZ 78°	8 [(a)]	0.88
W24x110 BEAM	El. 37'-3" Between AZ 42° 30' and AZ 620°	1 [(a)]	0.63
W33x200 BEAM	El. 37'-3" at AZ 139°	5 [(a)]	0.7
W33x200 BEAM	El. 37'-3" at AZ 270°	3 [(a)]	0.65
W24x110 BEAM	El. 37'-3" at AZ 247° 30'	8 [(a)]	0.88
W33x200 BEAM	El. 37'-3" at AZ 227°	8 [(a)]	0.76

TABLE 3.8.3-4 (Continued)

CONTAINMENT INTERNAL STRUCTURES SUMMARY OF GOVERNING COMBINED STRESS RATIOS FROM THE BEAM/COLUMN INTERACTION EQUATION FOR PRINCIPAL STRUCTURAL STEEL MEMBERS

Description of Principal Members	Location of Principal Members	Governing Load Combination Number	Combined Stress Ratio (<1.0)
W24x110 W/WT8x48 BEAM	El. 37'-3" Between AZ 247° 30' and AZ 270°	1 [(a)]	1.0 [(b)]
W24x84 W/PL 1"x8" BEAM	El. 37'-3" Between AZ 270° and AZ 284°30'	1 [(a)]	1.0 [(b)]
W33x141 BEAM	El. 52'-0" at AZ 25° 30'	8 [(a)]	1.0 [(c)]
W33x141 BEAM	El. 52'-0" at AZ 42° 30'	6 [(a)]	0.75
W33x200 BEAM	El. 52'-0" at AZ 62° 0'	1 [(a)]	0.52
W24x145 BEAM	El. 52'-0" Between AZ 5° and AZ 25° 30'	1 [(a)]	0.43
W33x200 BEAM	El. 52'-0" at AZ 90°	8 [(a)]	0.99 [(c)]
W33x200 BEAM	El. 52'-0" at AZ 106° 30'	8 [(a)]	0.8
W33x141 BEAM	El. 52'-0" at AZ 106° 30'	8 [(a)]	0.91 [(c)]
W14x103 BEAM	El. 52'-0" Between AZ 106° 30' and AZ 127°	8 [(a)]	1.08 [(c)]
W30x210 W/1"x8" COVER AT T&B/FLG.	El. 52'-0" Between AZ 247° 33' and AZ 270°	8 [(a)]	0.84
W30x190 BEAM	El. 52'-0" Between AZ 247° 33' and AZ 270°	8 [(a)]	0.75
W30x190 BEAM	El. 52'-0" Between AZ 247° 33' and AZ 270°	8 [(a)]	0.6

TABLE 3.8.3-4 (Continued)

CONTAINMENT INTERNAL STRUCTURES SUMMARY OF GOVERNING COMBINED STRESS RATIOS FROM THE BEAM/COLUMN INTERACTION EQUATION FOR PRINCIPAL STRUCTURAL STEEL MEMBERS

Description of Principal Members	Location of Principal Members	Governing Load Combination Number	Combined Stress Ratio (<1.0)
W33x141 BEAM	El. 52'-0" at AZ 227°	8 [a]	0.61
W30x172 W/½x12" PL AT T&B FLG. BEAM	El. 52'-0" Between AZ 247° 33' and AZ 270°	8 [a]	0.97 [c]
W24x145 BEAM	El. 52'-0" Between AZ 334° 30' and AZ 355°	8 [a]	0.57
W24x145 BEAM	El. 52'-0" Between AZ 302° and AZ 322° 30'	8 [a]	0.74
W30x190 BEAM	El. 52'-0" Between AZ 284° 30' and AZ 302°	8 [a]	0.91
W30x116 BEAM	El. 52'-0" Between AZ 270° and AZ 284° 30'	8 [a]	0.88
W24x120 BEAM	El. 52'-0" Between AZ 270° and AZ 284° 30'	8 [a]	0.74
W30x116 W/1/2x12" PL AT T&B FLG. BEAM	El. 52'-0" Between AZ 270° and AZ 284° 30'	8 [a]	0.87
W18x50 BEAM	El. 52'-0" Between AZ 284° 30' and AZ 302°	6 [a]	0.89
W33x200 BEAM	El. 52'-0" at AZ 247° 33'	8 [a]	0.94
W33x152 W/3/4"x8 " PL AT T/FLG. BEAM	El. 19'-0" at AZ 78°	5 [a]	0.73
W33x152 W/3/4 x 8" PL AT T/FLG. BEAM	El. 19'-0" Between AZ 169° and AZ 180°	6 [a]	0.91
W24x100 BEAM	El. 19'-0" at AZ 284° 30'	8 [a]	0.95 [c]

TABLE 3.8.3-4 (Continued)

CONTAINMENT INTERNAL STRUCTURES SUMMARY OF GOVERNING COMBINED STRESS RATIOS FROM THE BEAM/COLUMN INTERACTION EQUATION FOR PRINCIPAL STRUCTURAL STEEL MEMBERS

Description of Principal Members	Location of Principal Members	Governing Load Combination Number	Combined Stress Ratio (<1.0)
W16x40 BEAM	El. 19'-0" Between AZ 270° at AZ 284° 30'	8 [a]	0.85
W24x100 BEAM	El. 19'-0" at AZ 247° 30'	8 [a]	1.04 [c]
W33x152 BEAM	El. 19'-0" at AZ 247° 30'	8 [a]	0.68
W24x100 BEAM	El. 19'-0" Between AZ 227° and AZ 247° 30'	4 [a]	0.71
W24x120 BEAM	El. 19'-0" Between AZ 227° and AZ 247° 30'	8 [a]	0.88
W14x176 BEAM	El-2'-0" at AZ 98° 15'	8 [a]	0.66
W24x160 BEAM	El-2'-0" Between AZ 180° and AZ 206° 30'	5 [a]	0.45
W24x94 BEAM	El-2'-0" Between AZ 180° and AZ 206° 30'	1 [a]	0.48
W24x130 BEAM	El-2'-0" Between AZ 180° and AZ 206° 30'	1 [a]	0.48
W24x160 BEAM	El-2'-0" Between AZ 227° and AZ 247° 30'	5 [a]	0.53
W12x22 W/3/8"x6" PL AT T&B FLG.	El-2'-0" at AZ 270°	8 [a]	0.84
W24x130 BEAM	El-2'-0" Between AZ 180° and AZ 206° 30'	8 [a]	0.38

a. Refer to Table 3.8.3-2 for description of load combination number.
b. Based on L.L. = 200 PSF
c. The elastic section modulus of steel shape was used in the load combination. The plastic section modulus instead of elastic section modulus of steel shape may be used in this load combination.

TABLE 3.8.4-1

LOAD COMBINATIONS FOR CONCRETE CATEGORY I STRUCTURES [b,c]

LOADS

Category	Loading Condition	No.	D	L	T_o	R_o	E_o	W	E_{ss}	W_t[f]	H	T_a	R_a	P_a	Y[f]	Strength (See Notes)
Service	Normal	1	1.4	1.7												U for All Combinations (Note a)
		2[d]	1.4	1.7	1.7	1.7										
	Severe Environmental	3	1.4	1.7			1.9									
		4	1.4	1.7				1.7								
		5[d]	1.4	1.7	1.7	1.7	1.9									
		6[d]	1.4	1.7	1.7	1.7		1.7								
		7	1.2				1.9									
		8	1.2					1.7								
Non-Service	Abnormal	9[e]	1.0	1.0	1.0	1.0						1.0	1.0	1.5		
	Abnormal/Severe Environmental	10[e]	1.0	1.0	1.0	1.0	1.25					1.0	1.0	1.25	1.0	
	Extreme Environmental	11	1.0	1.0	1.0	1.0			1.0							
		12	1.0	1.0	1.0	1.0			1.0							
		13	1.0	1.0	1.0	1.0					1.0					
	Abnormal/Extreme Environmental	14[e]	1.0	1.0	1.0	1.0			1.0			1.0	1.0	1.0	1.0	

TABLE 3.8.4-1 (Continued)

LOAD COMBINATIONS FOR CONCRETE CATEGORY I STRUCTURES [b,c]

NOTES:

a. "U" is the section strength required to resist design loads and is based on methods described in ACI 318-71.

b. Loads not applicable to a particular system under consideration may be deleted.

c. If the effect of a nonpermanent load reduces the effect of others in the combination, the case of it being absent shall also be considered.

d. The values of load factors in these combinations are multiplied by a factor of 0.75 to account for T_o and R_o in addition to factors in the table.

e. In combinations 9, 10, and 14 the maximum values of P_a, $(T_o + T_a)$, $(R_o + R_a)$ and Y including an appropriate dynamic load factor shall be used unless a time history analysis is performed to justify otherwise.

f. In determining appropriate loads for Y or tornado missiles elasto-plastic behavior may be assumed with a maximum ductility ratio as stated in Table 3.5-13, provided excessive deflections do not result in loss of function of any safety-related system.

TABLE 3.8.4-2

LOAD COMBINATIONS FOR STEEL CATEGORY I STRUCTURES [a,b]

LOAD

Category	Loading Condition	No.	D	L	T_o	R_o	E_o	W	E_{ss}	W_t [f]	H	T_a	R_a	P_a	Y [f]	Strength (See Note c)
Service	Normal	1	1.0	1.0												S
		2	1.0	1.0	1.0	1.0										1.33S
		3	1.0	1.0			1.0									S
Severe Environmental		4	1.0	1.0				1.0								S
		5	1.0	1.0	1.0	1.0	1.0									1.33S
		6	1.0	1.0	1.0	1.0		1.0								1.33S
Non-Service	Abnormal	7 [e]	1.0	1.0	1.0	1.0						1.0	1.0	1.0		1.60S
	Abnormal/Severe Environmental	8 [e]	1.0	1.0	1.0	1.0	1.0					1.0	1.0	1.0	1.0	1.60S [d]
		9	1.0	1.0	1.0	1.0			1.0							1.60S
	Extreme Environmental	10	1.0	1.0	1.0	1.0				1.0						1.60S
		11	1.0	1.0	1.0	1.0					1.0					1.60S
	Abnormal/Extreme Environmental	12 [e]	1.0	1.0	1.0	1.0			1.0			1.0	1.0	1.0	1.0	1.70S [d]

TABLE 3.8.4-2

LOAD COMBINATIONS FOR STEEL CATEGORY I STRUCTURES [a,b]

NOTES:

a. Loads not applicable to a particular system under consideration may be deleted.

b. If the effect of a nonpermanent load reduces the effect of others in combination, the case of it being absent shall also be considered.

c. "S" is the required strength based on the elastic design methods and allowable stresses defined in Part I of the AISC Specification.

d. For these two combinations, the plastic section modulus of steel shapes may be used in computing the required strength, "S".

e. In combinations 7, 8, and 12, the maximum values of P_a, $(T_o + T_a)$, $(R_o + R_a)$ and Y including an appropriate dynamic load factor shall be used unless a time history analysis is performed to justify otherwise.

f. In determining appropriate loads for Y or tornado missiles, elasto-plastic behavior may be assumed with a maximum ductility ratio as stated in Table 3.5-13, provided excessive deflections do not result in loss of function of any safety-related system.

APPENDIX 3.8.A

COMPUTER PROGRAMS USED IN STRUCTURAL ANALYSIS

3.8.A.1 Computer Programs

The following are the abstracts of the computer programs used for the design of plant structures. The accuracy of the programs has been validated by comparison with results from manual calculations and/or commercially available computer programs. Areas of applications are shown in Table 3.8.A-1.

3.8.A.2 Static Analysis of Thin Shells of Revolution (ES418)

This program is based on the publication, "Static, Free Vibration and Stability Analysis of Thin, Elastic Shells of Revolution," by A. Kalnins, Technical Report AFFDL-TR-68-144, Wright-Patterson Air Force Base. The program has been modified to include the capability to analyze shells continuous across the axis of revolution and shells supported by elastic springs.

The program is based on the classical theory of thin elastic shells. Nouniform loads in an axisymmetrical shell are transformed into a Fourier expansion series in the circumferential (hoop) direction. Thus, the governing eight partial differential equations can be transformed into eight first-order, linear, ordinary differential equations. The method used to solve these equations is a direct integration technique based on Runge-Kutta method. The boundary value problem of the shell is then treated as an initial value problem. The direct integration starts at the first segment, where four boundary values are prescribed.

After the initial value problems are integrated over the successive segments, continuity conditions on all variables are written at the end points of the segments, constituting a system of simultaneous, linear matrix equations. This system of matrix equations is then solved directly by means of Gaussian elimination. The computer program can handle any arbitrary shell of revolution. Loadings may vary nonuniformly in the meridional and circumferential directions, and can be mechanical or thermal loads. The mechanical loads may be distributed and/or discrete loads. The shell members may have several layers with different isotopic and/or orthotropic elastic properties. The thickness of layers may vary along the meridian. According to its shape, sectional thickness, material properties, and loading types, a shell may be divided into 20 parts. Each part is identified as a certain type of shell, such as a cylindrical, spherical, toroidal, conical, paraboloidal, ellipsoidal, hyperbolical, or a general shell. The shell may be supported by elastic springs, representing a condition such as the Containment mat resting on soil. The boundary condition may be deformations and/or forces. In the case of shells continuous across the axis of symmetry, only the initial boundary values are required.

As output, this program furnishes the deformations (displacements and rotations), membrane forces (meridional and circumferential axial forces and in-plane shear forces), bending moments (meridional and circumferential), and radial shear forces. In addition, there is an option to have stresses printed out in the extreme fibers of each section.

3.8.A.3 NASTRAN

NASTRAN (NASA Structural Analysis) was developed under the sponsorship of the National Aeronautics and Space Administration (NASA) by a committee with representation from eight NASA centers (for specifications), and by the Computer Sciences Corporation and Bell Aerosystems Company (for the implementation). NASTRAN is a finite element computer program for structural analysis that is intended for general use. Structural elements include rods, beams, shear panels, plates, shells of revolution, and scalar and solid polyhedron elements. The range of analysis types in the program includes static response to concentrated and distributed loads, to thermal expansion, and to enforced deformation; dynamic response to transient loads, to steady-state sinusoidal loads, and to random excitation; and determination of real and complex eigenvalues for use in vibration analysis, dynamic stability analysis, and elastic stability analysis. The program also includes a limited capability for the solution of nonlinear problems, including piecewise linear analysis of nonlinear static response and transient analysis of nonlinear dynamic response.

The displacement method has been employed throughout the analysis. Structures are modeled with finite elements, including plate elements, shell of revolution elements, shear panels, beams, and rods. Elements are identified by numbers, and are interconnected at a finite number of grid points. The grid points may be defined by a basic or local coordinate system. Each grid point may have 6 degrees of freedom, representing three displacements and three rotations. After receiving the input data, the first task of the program is to generate the stiffness matrix and the load vector. The next step is the matrix decomposition, which is especially important because of the required computing time, possible error accumulation, and numerical instability. The program takes maximum advantage of matrix sparsity and bandedness. The band width is influenced greatly by the user, who establishes the numbering system for the grid points.

Using the finite element technique, any type of structure can be accurately modeled. Deformation constraints (displacements and rotations) may be imposed on any grid point. Boundary conditions may be imposed on any grid point. Boundary conditions may be homogeneous or nonhomogeneous. As output from the analysis, the displacements and rotations for each grid point and the moments and stresses in each element are printed out. Forces in elements may be calculated through the output stresses at the two extreme fibers of each element.

3.8.A.4 Static Analysis of Shells – Data Management (ES420)

This Prestressing Tendon Force analysis Program was developed to analyze U-shaped tendon systems for Containment structures. The tendons run continuously up the cylindrical wall, over the dome, and down the other side of the wall, and are anchored at both ends at the same level below the foundation mat inside a tendon gallery. As input, the number of tendons must be given. Also, the Containment dome must be divided into a sufficient number of parts by colatitude and meridional lines. The program identifies the tendons associated with each respective part. Forces created by tendon curvature friction, and by jacking and lockoff post-tensioning forces, are calculated for each tendon and are combined for each part. These forces consist of radial, colatitude, and meridional forces, which are converted to unit forces on each part. The last part of the analysis is to transform the unit forces into a Fourier expansion series, which forms the load data for the general shell analysis.

3.8.A.5 Interaction Analysis for Members Subjected to Bending With Axial Load and Thermal Gradient (ES423-WSD and ES424-USD)

This program consists of interaction analysis for working stress design and for factored load design. The analysis includes the effect of thermal gradient and considers variable locations of axial forces.

The basic assumptions of the program are as follows: a plane section remains a plane section after bending; the tensile capacity of concrete is neglected, the temperature varies linearly across the concrete section; the materials respond perfectly elastic below yield, and thereafter perfectly plastic; and for working stress design, allowable stresses must be specified, while for factored load design, maximum allowable strains must be specified along with the yield stresses.

The axial-force-versus-bending-moment-interaction diagrams are plotted by computer-controlled plotters, such as Calcomp, with reinforcement ratios and allowable stresses or allowable strains as parameters. The axial forces and bending moments are expressed in terms of sectional areas and stress units, and thus appear in a dimensionless form.

One important option of the program is that the location of the axial forces need not be specified when the axial forces act at the centroid of a cracked section. In such a case, the centroid is calculated first and is assigned to be the location of the axial forces.

3.8.A.6 ICES STRUDL – II – The Structural Design Language

This program was developed at the Massachusetts Institute of Technology (MIT) Civil Engineering Systems Laboratory. Analytic procedures in ICES STRUDL – II apply to both framed structures and continuous mechanical problems. Framed structures are either two- or three-dimensional structures composed of slender, linear members, which can be represented by properties along a centroidal axis. Such a structure is composed of joints, including support joints and members connecting the joints. Continuous mechanical problems are treated, in STRUDL, using the finite element method. In this method, the domain of the problem is sub-divided into one-, two-, or three-dimensional elements of different shapes, and connected at a finite number of nodal points or joints. STRUDL provides a variety of element types for the solution of plane stress/strain, plate bending, and shell analysis problems. Mixing of different element types is allowed in the program.

The analytic procedures presently available in ICES STRUDL are determinate analysis, preliminary analysis, stiffness analysis, nonlinear analysis, linear buckling analysis, and dynamic analysis. Results for member end forces and distortions, reactions and joint displacements, and element stresses and strains are provided. The determinate analysis is the solution by statics alone of a framed structure acted upon by a series of loading conditions. The preliminary analysis procedure is applied to an indeterminate problem for which the engineer has provided sufficient information or assumptions on structural behavior under each loading condition to render the problem statically determinate. The stiffness analysis is a linear, elastic, static, small displacement analysis applying a procedure which requires the specification of member and element properties in some acceptable form and treats the joint displacements as unknowns. In the nonlinear analysis the geometric nonlinearity is treated by formulating the set of nonlinear equations, using a nonlinear strain displacement relationship. These equations may be solved by iterative techniques, using either the Newton-Raphson or the successive iteration technique. For the linear buckling procedure, nonlinear terms in the formulation are dropped, and the problem is reduced to an eigenvalue problem. This problem is solved by using the Stodola Vianello method, leading to the lowest buckling load factor corresponding to the applied load pattern. The dynamic analysis capabilities include the solution of the real eigenvalue problem and the determination of the displacement and force response. The solution of the eigenvalue problem makes use of the Householder-Sturm-Ortega method. The determination of response may be made by modal superposition, or by direct integration of the equations of motion, in which case a linear acceleration method is used. The results of the response analysis are the dynamic displacements and forces.

3.8.A.7 Cross-Sectional Properties and Weight System Resultant (ES415)

This program performs calculations to determine the property values of a cross-section composed of simple geometric elements. Computed values include the total cross-sectional area, weight per linear unit, elastic centroid, center of gravity, moment of inertia about the major axis, mass moment of inertia about an arbitrary point, section moduli, bending stress, and the resultant force vector. The required input data for the above calculations are the geometric description of the cross-sectional area, the weight per unit volume, and reference points for the computation of section moduli and of the mass moment of inertia.

3.8.A.8 Buckling of Shell of Revolution (BOSOR 4)

This program was developed by Lockheed Missiles and Space Co. under the sponsorship of Naval Ship Research and Development Center. The program was developed for general axisymmetric shell structure including segmented, ring-stiffened, and branched shells of revolution. The shell is input in a maximum of 25 segments and branches specifying a maximum of 450 mesh points. The program can accommodate 50 discrete circumferential rings and any number of smeared rings and meridional stiffeners.

The program has the following capabilities:

1. Stress and displacement analysis from a nonlinear theory for stepwise increasing mechanical and/or thermal loads.

2. Critical buckling load for axisymmetric collapse.

3. Buckling load for nonaxisymmetric buckling modes for a range of circumferential wave numbers, seeking the minimum load.

4. Vibration analysis of shells corresponding to axisymmetric and nonaxisymmetric modes.

5. Stress, displacement and buckling analysis of nonsymmetric loaded shells.

The output includes the meridional and circumferential stresses or stress resultants, thermal stress resultants, meridional rotations, meridional, circumferential and twist moment resultants, and the reference surface displacements. The plots are optional.

3.8.A.9 GERMIS and GEMP – Geotechnical information has been analyzed and stored using both GEMIS and GEMP

3.8.A.9.1 GEMIS – Geotechnical Engineering Monitoring Information System (ES211). GEMIS provides analysis and storage of geotechnical information that varies as a function of time. The geotechnical information is vertical movement of soil strata, position of groundwater table, progress of excavation, and pore pressures.

All geotechnical information is expressed as a relative change from the value at day-zero (beginning of measurement period). These relative changes are stored with the associated time of occurrence for later use, in the form of tables and graphs.

Subsequent to Architect-Engineer transition (Brown & Root to Bechtel) in early 1982 GEMIS was replaced by GEMP, described in Section 3.8.A.9.2.

3.8.A.9.2 GEMP – Geotechnical Engineering Monitoring Program. GEMP is conceptually similar to GEMIS for the processing and storage of geotechnical data and is compatible with GEMIS conventions. All stored GEMIS data has been merged into the file base so that there is historical continuity of information.

Presentation of output from stored data is, in principal, similar to GEMIS output. However, formats in some instances are different or have evolved in response to the continued reviews of foundation behavior and groundwater performance.

3.8.A.10 Plane Section Properties and Shear Force Distribution on an Assemblage of Rectangular Sections (ES432)

This program provides the user with the various sectional properties of a shear wall assembly required to model a structure for dynamic analysis. Required input consists of the location and dimensions of each wall element and various data defining the connectivity of these elements. At present, only rectangular elements will be considered, and any number of closed or open sections can be included. Output consists of section properties and the shear force distribution due to a 100-kip force in the X and Y directions and a torque of 100,000 K-ft. Shear areas, torsional constant, and shear force distribution are calculated based on either the thin wall theory or rigidity approach.

3.8.A.11 Determination of Stress and Displacements in an Elastic Half-Space (ES213)

This program calculates the stresses and displacements at specified points of an isotropic, homogeneous, elastic half-space subjected to any arbitrary loading.

The program is used in the geotechnical analysis for safety-related structures.

3.8.A.12 SETTLE

The program SETTLE calculates settlement using the one-dimensional Terzaghi consolidation theory and equations. For the South Texas Project Electric Generating Station (STPEGS) settlement evaluations SETTLE used the H-SPACE (Section 3.8.A.11 above) output to calculate the consolidation (settlement).

3.8.A.13 NASTRAN Data Management – "Penetration Tendon Analysis" (ES428)

This program calculates the forces and moments to be applied at the nodes of a finite element mesh due to the prestressing tendons. Each element of the finite element mesh is examined and the tendons passing through it are identified. The resulting forces are considered to act at the midpoint between the points of intersection.

The maximum number of critical tendon points allowed is nine. If a buttress falls within the penetration model, three additional critical points will be generated by the program to give a maximum of twelve.

3.8.A.14 Fourier Series Analyses (ES017)

The purpose of this program is to calculated the coefficients in a Fourier Series Expansion which best describes the data input. The entered data must be periodic in the interval 0 to 2. The periodic characteristics inherent in the trigonometric functions describing a Fourier Series allow previous input or calculated coefficients to be either multiplied or divided by an integral multiple of the period. The program has the option to check the fit by plotting the input values and the calculated values of the expansion. The program is capable of plotting either the input points or the Fourier function of the entered coefficients.

3.8.A.15 Generation of New Time History Load (ES434)

The program is used to process multiple time history load data, compute and punch new time histories in format suitable for input to ICES STRUDL DYNAL program. The program is capable of processing a maximum of five time histories, averaging the input data to obtain the new time histories.

3.8.A.16 SRSS Loading Combination (ES439)

This program is used to process results from a finite element analysis using computer programs such as NASTRAN. The program converts normal stresses to normal forces and then combines these forces obtained from various loading conditions using the Square Root of the Sum of the Squares (SRSS) technique.

In addition, the program can also perform simple load combinations utilizing a multiplicative on the calculated normal forces, bending moments and shears input by the user.

The maximum number of finite elements allowed is 450, and the maximum subcases that may be used in a loading combination is 12.

3.8.A.17 ICES COGO

ICES COGO (COordinate GeOmetry) is an information processor for the computer solution of geometric problems in civil engineering, site planning, and related areas. The processor is made up of a language, a set of processing routines, and information files. While COGO language is designed for the natural expression of geometric problems in the civil engineering areas, it can be used for a variety of geometric problems in two or three dimensional space.

This program was developed by the Civil Engineering System Laboratory at MIT.

3.8.A.18 Slope Stability Analysis System (SLOPE)

SLOPE is an engineering computer program that offers advanced techniques for the automated analysis of slope and embankment stability problems. SLOPE is also an ICES subsystem and as such offers the facility of a problem-oriented program language built upon soils engineering terminology.

SLOPE contains the following methods of stability analysis:

Bishop Method, whose equilibrium forces are resultant horizontally to the side of the slope slices, estimates the factor of safety against failure along a circular failure arc;

Fellenius Method, whose equilibrium forces are resultant parallel to the bottom of the slope slices, estimates the factor of safety against failure along a circular failure arc;

Morgenstern and Price Method, whose equilibrium forces are calculated by the use of non-linear equation, estimates the factor of safety against failure along any arbitrary failure surface.

Input data is comprised of slope geometry, soil profile, soil properties, and water condition commands. The SLOPE subsystem also provides the user with the capability to introduce earthquake loading.

SLOPE was developed at the MIT. McDonnell Douglas Automation Company (MCAUTO) offers this program through their facilities.

3.8.A.19 STPSYS – A System to Calculate Settlements (ES214)

STPSYS is a system of four programs linked by files for the purpose of calculating the magnitude and progress of settlement in a multi-strata, multilayer soil system. The program utilizes the finite difference method to solve the problem of one-dimensional soil consolidation. The four programs were originally developed by R.L. Schiffman of the University of Colorado as stand-alone programs.

3.8.A.20 Response Envelope (ES431)

This computer program performs log and multiplication transformation of data, generates envelopes of shifted curves and produces two-dimensional plot output. This program was primarily used to generate floor design response spectra from the time history response of primary structures. The envelopes are used in the design of subsystems such as equipment, piping etc. This program can also be used for some two-dimensional plots.

3.8.A.21 ICES STRUDL DYNAL

ICES STRUDL DYNAL is the result of a merger of ICES STRUDL and the predecessor dynamic analysis system, DYNAL. The merger incorporates DYNAL's analysis capabilities into STRUDL while greatly expanding the output facilities. STRUDL DYNAL has been developed through the efforts of McDonnell Douglas Automation Company and Engineering Computer International and maintained at McDonnell Douglas Automation Company, St. Louis, Missouri.

ICES STRUDL DYNAL uses the modal superposition method to obtain system responses of a structure. The program generates the mass and stiffness matrices representing the distributed mass and stiffness of the actual structure and then performs the modal analysis to obtain the mode shapes and natural frequencies of the system. The frequencies are the eigenvalues and the mode shapes, the eigenvectors of the eigenvalue-eigenvector problem. These results are used to obtain the response of the structure to a specified form of the excitation. The forms of excitation available are Shock Spectrum, Harmonic, and Transient.

3.8.A.22 STRUCTURAL WELDS (SP-291 AO)

Computer Program WELD is used to calculate stresses in a fillet weld of equal leg size. The program is capable of handling weld groups with various types of defects. Defects in the welds reduce weld section properties and could shift the centroid of the weld group. The shifting of the centroid of the weld group causes and additional twisting moment from applied concentric eccentric forces. This would result in a reduction in the load carrying capacity of the weld group. The program calculates maximum stress in a weld group from a set of three-dimensional input forces applied at any point away from the origin of the weld group. In addition, the capacity of the defective weld under vertical load is obtained.

3.8.A.23 STARDYNE

STARDYNE is a finite element program for static and dynamic structural analysis. A STARDYNE static analysis will predict the stresses and deflections resulting from pressure, temperature, concentrated forces, and enforced displacements. Dynamic analysis will predict the node displacements, velocities, accelerations, element forces, and stresses from transient, harmonic, random, or shock excitations. STARDYNE is user-oriented, containing automatic node and element generation features that reduce the effort required to generate input. Plots of the original model and deformed structural shapes help the user to evaluate results. Contour plots show surface stress for two-dimensional elements. The program creates time histories of element forces and stresses and of node displacements, velocities, and accelerations.

This program was developed by System Development Corporation, 2500 Colorado Avenue, Santa Monica, California, 90406 and is maintained by Control Data Corporation (CDC).

STARDYNE may be accessed worldwide through service centers in major cities or via private remote batch or time-sharing terminals.

3.8.A.24 BASEPLT and BASEPLATE II

Program BASEPLT and BASEPLATE II generate the finite element input model of flexible baseplate for the STARDYNE analysis. The programs require a minimum amount of input to define the problem. The square or rectangular baseplate of uniform thickness is represented by quadrilateral elements (QUADB). The baseplate is attached to the support by randomly located bolts with tension and shear stiffness only. The supporting concrete modeled as an elastic foundation consists of compression springs connecting the baseplate to the fixed support. The baseplate is stiffened by various types of attachment; forces and moments are applied at either the top or bottom of the attachment. STARDYNE obtains the solution of the non-linear baseplate problem by substructure technique. The post-processor of BASEPLT and BASEPLATE II condenses STARDYNE output into a minimum amount of information necessary for the design of baseplate and anchor bolts.

3.8.A.25 BSAP (CE800) – Bechtel Structural Analysis Program

The Bechtel Structural Analysis Program (BSAP) is a general purpose finite-element computer program for analysis of structural systems subject to static, dynamic, and thermal loads. The program incorporates and extensive library of beam, shell, and solid elements, such that virtually any type of structure can be represented. Common applications include analysis of nuclear plant structures, pressure vessels, high rise buildings, transmission towers, and bridges. BSAP is based upon and incorporates features of the SAP program developed at the University of California, Berkeley by Professor E. L. Wilson, plus modifications to extend the capabilities, enhance the usability, and to reduce the cost of application.

Static loads that may be considered include nodal forces, distributed pressures, differential temperatures, and boundary movements. The static solution is obtained using the Gaussian elimination technique or the Crout elimination technique. Modal extraction may be carried out on a dynamically condensed system using the Householder-QR algorithm, Ritz reduction and Jacobi method, or on the full system using Subspace Iteration or Determinant Search methods. Dynamic analysis capabilities include seismic response spectrum or time history, time-history forces and steady-state frequency response. Multiple static load cases for a given structural model may be analyzed in a single computer run.

Use of the program requires development of a finite-element analytical model of the structure under consideration. In the model, the structure is represented as an assemblage of finite elements interconnecting discrete nodal points. Information input to the program includes geometric configuration, element data, material properties, and specification of loads.

3.8.A.26 BSAP-POST (CE201/CE217)

BSAP-POST is a general purpose post-processor program for the Bechtel Structural Analysis Program. Its processing capabilities include the following modules:

3.8.A.26.1 UTILITY Module. The primary task for the UTILITY Module is to retrieve or save, in a sequential format, the BSAP-POST data base. It can also merge two or more data bases into a single data base. The five parts of the data base (MODEL, VECTOR, STR, PLTFRC, RSAMD) can be processed separately. Each part is a single sequential file when stored on magnetic tape. The input data for UTILITY consists of a set of card images and a BSAP-POST data base or set of data bases. Each card image contains a keyword that refers to a specific operation that is related to the retrieval or saving of the BSAP-POST data base.

3.8.A.26.2 PLOT 3D Module. The purpose of the POLT 3D module is to plot undeformed and deformed geometries of structural models analyzed with BSAP. The deformed geometries may be static displaced shapes or mode shapes. The modules has a variety of options which give the user flexibility in defining the geometries that are to be plotted.

3.8.A.26.3 COMBINE Module. The primary task performed by COMBINE is to form new (factored) load case data by combining old (primary) load case data. STR, VECTOR, and PLTFRC type data can be processed. Scalar combinations of load case data can be formed using subroutines supplied with BSAP-POST. Nonscalar combinations of load case data can be formed using user written subroutines.

3.8.A.26.4 OPTCON Module. The OPTCON computer code described herein was developed to be a versatile and complete design and analysis program for reinforced structures. It assumes that section forces are available from analysis of the structure using other means (e.g., BSAP, FINEL, ASHSD, hand calculations, etc.). It can be used for determining the reinforcing steel requirements for a prestressed concrete section so long as the prestressing force is available and is treated as a section force. The program can be used for the investigation of an existing section where the reinforcing steel area is given. Or it can be used for obtaining an optimum design by letting the program determine the minimum reinforcement required.

The OPTCON program was originally intended for the design of containment structures for nuclear power plants. Thus, two of the three design methods (i.e., design for Service Loads and design for Factored Loads) are developed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) code, Section III, Div. 2. However, a third design method, that of design in accordance with the ultimate strength provisions of the ACI 318-71 code is also provided so that the program can be used for other structures. Furthermore, a special load combination subroutine is incorporated to facilitate the use of the program for any loading condition(s).

The design of nuclear power plant components usually requires consideration of thermal effects. For this reason, special subroutines are provided to incorporate the thermal effects into the design and/or investigation. The cracking effect of the concrete and yielding effects of the reinforcement (as allowed by the appropriate stress/strain/yielding criteria) are considered in the calculation of the thermal loads and moments computed by the program.

The program has the capability of considering a "hot" liner plate on one face of the concrete of the section. Any temperature may be applied to the liner plate (including a zero temperature effect) for any loading combination. Yielding of the liner plate, if it occurs, is considered.

3.8.A.26.5 PRINT Module. The purpose of the PRINT module is to print the contents of the BSAP-POST data base. The module has options that allow the user to selectively choose portions of the data base to be printed.

3.8.A.26.6 3DPLOT Module. The purpose of the 3DPLOT module is to plot contour plots of displacements and element stresses of structures (models) analyzed with BSAP. This module has a variety of options which give the user flexibility in defining the geometrics that are to be plotted.

3.8.A.26.7 RESULT Module. The continuum element force/stress output from BSAP is for each individual element. In many cases, the required output is a resultant force or moment due to the collective action of several individual elements. The RESULT module is intended to compute these resultant forces and moments. This is accomplished by defining a grid element that represents a stack of elements. This stack of elements is composed of BRICK and MEMB elements. The resultant forces and moments due to the elements in the stack can be oriented in a user defined coordinate system. These resultant forces and moments can be saved on the BSAP-POST data base and processed by other modules such as OPTCON and PRINT.

3.8.A.26.8 SELECT Module. The SELECT module can be used to extract selected vector components and write them to a file that can be used to interface with other programs. For example, SELECT can be used to extract the components of a flexibility matrix that was created by applying unit loads on a large finite element model of a nuclear reactor support structure.

3.8.A.26.9 PLOT TH Module. The PLOT TH module of BSAP-POST is a time history plotting processor. With PLOT TH the user can plot any variable (which is a function of time) versus time. The time histories to be plotted can be input in card form as equal or unequal interval time histories. PLOT TH can also be used to generate plots of the time histories generated by BSAP that exist on the time history post-processing tapes (TAPE11 and TAPE15). A standard tape input and output capability is also available. The time history data can also be modified by shifting, scaling, baseline correcting, and integrating.

3.8.A.27 BSAP-DYNAM (CE207)

BSAP-DYNAM (CE207) is one of several post-processors available in the BSAP analysis system. BSAP-DYNAM consists of a number modules that provide additional calculating capabilities or data interfaces to other Bechtel computer programs. These capabilities are primarily oriented toward the analysis of soil-structure interaction problems. The data interface between BSAP (CE800) and BSAP-DYNAM (CE207) is the Checkpoint-Restart file. The data interface between BSAP-POST (CE201) and BSAP-DYNAM (CE207) is TAPE27.

3.8.A.28 BSAP-PRE (CE212)

BSAP-PRE (CE212) is an interactive program operated by the user at a computer terminal (TTY or graphics terminal). It is used to create and edit input data decks for the BSAP (CE800) program in a manner that is simple and convenient, and it enables rapid data deck correction or modification. The finite element modal data can be obtained from data directly entered by the user, a file created by standard computer system editors, or from an analysis file (in Universal Format) from SUPERTAB (UE150).

BSAP-PRE is specifically designed for static and/or modal analysis of frame structures composed of BEAM, TRUSS, PIPE (straight and curved), SPRING, and BOUND elements. It is intended to economically facilitate creation of BSAP problem runs of small to moderate size.

An interactive graphics model display program is included for viewing models prior to running BSAP. This plotting package is compatible with Textronix 4014 and 4027 terminals.

3.8.A.29 SPECTRA (CE802) – Response Spectra Analysis

The SPECTRA program computes the response spectra from an acceleration record digitized at equal time intervals. These spectra are plots of the maximum response of a single degree of freedom oscillator over a range of values of its natural periods and percent of critical damping.

The numerical method for computing the spectral values is based on the exact analytical solution of the governing differential equation. It is assumed that the accelerogram varies lineally between the time-history points. The response spectra are then constructed by monitoring of the maximum values of response parameters of each step of integration. The computed spectra may then be widened to account for the effect of structural frequency variation. The response spectra due to the three directional ground excitations can be combined by the method of the square root of the sum of the squares.

3.8.A.30 ASHSD (CE803) – Axisymmetric Solids and Shells

ASHSD was originally designed by Ghosh and Wilson at the University of California, Berkeley, as a linear elastic finite element program. ASHSD employs an isoparametric thin shell element and a constant strain solid element to describe axisymmetric geometry. Non-axisymmetric loadings are handled by a separation of variables solution technique which employs partial Fourier series representation of the load. Analysis options include: dead load, static load, arbitrary dynamic load, horizontal and vertical earthquake. Response calculations include: direct integration time history analysis, modal superposition time history analysis, and response spectrum modal analysis.

3.8.A.31 ASHPOST/ASHCOMB (CE823)

ASHPOST/ASHCOMB is a post processing series of programs for the ASHSD (CE803) program. The ASHCOMB program can be used to scale and superimpose results obtained from various ASHSD runs and makes it possible to describe almost any loading condition on an axisymmetric structure. ASHPOST provides a plotting module (PLOT2D) which plots deformed and undeformed geometry, as well as stress and displacement contours. In addition, the section program is available for calculation of section resultants through any contiguous section of solid elements.

3.8.A.32 FINEL (CE801) – Finite Element Program for Cracking Analysis

FINEL is a two-dimensional, static, small displacement, bilinear-elastic finite element stress analysis computer program. FINEL's primary purpose is to perform plane or axisymmetric stress analysis of reinforced concrete structures. The program allows for concrete cracking and reinforcement yielding. Loading includes: concentrated pressure, displacement, thermal, inertial, and – for axisymmetric problems – centrifugal forces. In addition to reinforced concrete structures, FINEL can perform linear stress analysis on structures that can be modeled as plane stress, plane strain, or axisymmetric. Axially symmetric material properties, with the same or different properties normal to the plane, can be considered.

3.8.A.33 TENDON (CE239) – Prestressed Forces on a Hemispherical Dome

The dome tendon computer program calculates forces and pressures on a hemispherical dome of a prestressed three buttress concrete containment building, resulting from prestress by two orthogonal groups of vertical dome tendons and one group of horizontal hoop tendons. One group of vertical dome tendons is located in parallel, vertical planes normal to the x-axis; the other group is located in vertical planes normal to the y-axis; while the third group is located in horizontal planes normal to the z-axis. The vertical dome tendons (the first two groups) have equal areas and equal spacing measured along the springline. The hoop dome tendons have equal areas, but the spacing may be either constant or may vary linearly with the latitude. The hoop tendons extend from the springline into the dome region up to 45° latitude.

In the analysis, the dome is subdivided into a grid pattern specified by the user. The program calculates the total pressure due to tendon forces at each grid node in the radial direction, normal to the dome surface, and in the circumferential (hoop or azimuth) and meridional directions. Nodal forces in the hoop and meridional directions are calculated at each node point. The pressures and forces calculated by this program are intended for use as input to a finite element computer program to determine the stress distribution in the dome.

3.8.A.34 TURMIS (CE450)

The TURMIS computer was developed to compute the damage probability of a nuclear power plant subject to the impact of missiles generated by turbine blades or disks when they fail. The program calculates the damage probability for three types of targets which are: Type 1, cylindrical target; Type 2, rectangular target; Type 3, a cylinder with a dome on top. The program combines the damage probabilities of all targets to yield a total damage probability. The damage criteria is specified as scabbing or perforation of the concrete barrier which is predicted by a formula or formulas which can be selected as input.

The TURMIS program computes and prints conditional strike probability given missiles ejected from the turbine, P_2; the conditional damage probability given missiles strike the target, P_3; and the conditional damage probability given missiles ejected from the turbine, P_2P_3. The probability of missile occurrence which is designated as P_1 can be the input and the unconditional damage probability is calculated and designated as P_4.

3.8.A.35 SHAKE3 (CE915) – Vertical Shear in Horizontal Layers

The SHAKE3 program computes the responses in a system of homogeneous, visco-elastic layers of infinite horizontal extent subjected to vertically traveling shear waves. The program is based on the continuous solution of the wave quation adapted for use with transient motions through the Fast Fourier Transform algorithm. The nonlinearity of the shear modulus and damping is accounted for by the use of equivalent linear soil properties using an iterative procedure to obtain values for shear modulus and damping compatible with the effective strains in each layer.

SHAKE is a public domain program which has been in use for several years. The program was developed at the University of California, Berkeley, under the sponsorship of the National Science Foundation by Schnabel, Lysmer and Seed (1972) and was published as EERC Report 72-12 in December 1972 with subsequent modifications by Udaka and Lysmer in September 1973. These modifications mainly involved altering specific subroutines to decrease execution time by up to 50 percent, and resulted in only slight differences in the program results. In this form, SHAKE is on the Bechtel UNIVAC System as SHAKE3 (CE915).

3.8.A.36 FLUSH (CE988) – Soil-Structure Interaction Time History Analysis

The computer program FLUSH is for two-dimensional seismic time history response analysis of a finite element soil-structure interaction system using the frequency domain solution procedure. The program was developed at the University of California, Berkeley, and is an improved version of the previous LUSH program. The program name FLUSH stands for <u>F</u>ast-<u>LUSH</u>.

For general applications, the soil medium of the soil-structure system must be horizontally layered soil stratum overlaying a relatively rigid base rock. Either the horizontal or the vertical seismic input motion can be prescribed at any elevation within the soil foundation. The program then performs a free-field deconvolution analysis of the input motion, if necessary, to generate the base rock motion for the interaction analysis.

The nonlinearity of the shear modulus and damping is accounted for by the use of equivalent linear soil properties using an iterative procedure to obtain values for shear modulus and damping compatible with the effective shear strains.

There are three element types available in the program, namely, the beam element, isoparametric plane strain element and the void element. The boundary conditions considered in the program are fixed boundary, transmitting boundary and viscous boundary.

3.8.A.37 CLASSI (CE982) – Continuum Linear Analysis for Soil-Structure Interaction

The "Continuum Linear Analysis for Soil-Structure Interaction" program (CLASSI) consists of a specialized substructure method which is capable of maximizing the details in the structural model as well as those for the foundation-soil system. The goal of the substructure analysis is to determine the unknown foundation motion, which depends on three major sets of information: (1) the dynamic characteristics of the substructures; (2) the force-displacement relationship of the foundation-soil system, simply represented by the impedance matrix; and (3) the characteristics and amplitudes of the incident waves. The program is capable of analyzing any structure which can be represented by a standard finite element program. In addition, it can analyze the motions of an arbitrarily shaped flat, rigid foundation on a layered soil.

The response of a structure undergoing base excitation can be expanded in the mode shapes of a fixed-based structure. Therefore, any finite element program may be applied first to obtain a meaningful and detailed representation of the fixed-base structural characteristics, after which this information is condensed by CLASSI into an effective mass matrix, which serves as the entry to the substructure analysis from the superstructures.

In the foundation analysis, CLASSI calculates the foundation impedance matrix and the foundation input motion from a given incident wave using an integral equation method. This type of analysis is especially attractive for geotechnical problems because the dimensions of the soil medium are extremely large compared to the length of the seismic waves. Furthermore, since the integral equations concern only the boundary values at the interface between the foundations and the soil medium, the computation is considerably reduced from the "volume type" formulation used by finite element or finite difference methods.

TABLE 3.8.A-1

COMPUTER PROGRAMS AND AREAS OF APPLICATIONS

Computer Program	Area Used
Static Analysis of Thin Shells of Revolution (ES418)/ASHSD/ASHPOST	Analysis of Containment tendon gallery, mat and shell
NASTRAN/BSAP/FINEL	In Reactor Containment Building: mat foundation, personnel and auxiliary air lock region, equipment hatch region, primary and secondary shield walls, pressurizer slab
	In Mechanical-Electrical Auxiliaries Building: tendon access emergency shaft walls, isolation valve cubicle walls and slabs
Static Analysis of Shells-Data Management (ES420)/TENDON	Dome tendon force analysis and Containment shell analysis
Interaction Diagram for Members Subject to Bending with Axial Load and Thermal Gradient (ES423-WSD) and (ES424-USD) / OPTCON	Design of Containment shell internals and Containment mat
ICES STRUDL – II / BSAP	In Reactor Containment Building: steel frame analysis, refueling pool support system
	In Mechanical-Electrical Auxiliaries Building: mat analysis, some walls, and some slabs
	In Fuel-Handling Building: mat analysis and spent fuel pool walls, and slabs
	In Diesel-Generator Building: mat analysis
Fourier Series Analysis (ES017)	Used to prepare input data for ES418 in Containment mat analysis and Containment shell analysis

TABLE 3.8.A-1 (Continued)

COMPUTER PROGRAMS AND AREAS OF APPLICATIONS

Computer Program	Area Used
Generation of New Time History Load (ES434)	Process multiple time history load data used in the seismic analysis of Category I structures
SRSS Loading Combination (ES439)	Post-processing of NASTRAN output data
ICES COGO	Determine alignment, area, etc., of Essential Cooling Pond including embankment
Slope Stability Analysis System (SLOPE)	Slope stability analysis of Essential Cooling Pond Embankment
STPSYS – A System to Calculate Settlements (ES214)	To calculate movement of ground settlement
Response Envelope (ES431) / SPECTRA	To prepare floor response spectra
ICES STRUDL DYNAL/BSAP	Seismic analysis of Category I buildings
STARDYNE/BSAP	For any static or dynamic analysis in any building where its application applies
BASEPLATE II	Analysis of embedded plates
Buckling of Shells of Revolution (BOSOR 4)	Static and instability analysis of the Containment dome liner for construction loading. All static analysis of Containment dome liner.
Geotechnical Engineering Monitoring Information System (GEMIS) (ES 211)	Processing, storage, and retrieval of data from the geotechnical instrumentation monitoring of ground, ground-water fluctuation, and foundation movements during construction.
Plane Section Properties and Shear Force Distribution on Assembly of Rectangular Section (ES 432)	Lumped-mass modeling of Category I buildings for seismic analysis.
Determination of Stress and Displacement in an Elastic Half-Space (ES 213)	Analysis of stress under Category I buildings due to changed loading condition.

TABLE 3.8.A-1 (Continued)

COMPUTER PROGRAMS AND AREAS OF APPLICATIONS

Computer Program	Area Used
NASTRAN Data Management-Penetration Tendon Analysis (ES 428)	Force distribution due to curved tendons around personnel and auxiliary air lock, and equipment hatch.
Cross-sectional Properties and Weight System Resultant of (ES 415)	Computation of Section Properties and Weight for various buildings.
SHAKE3 / FLUSH / CLASSI	Soil-structure interaction analysis for Category I structures
TURMIS	To compute the damage probability of buildings subject to the missiles ejected from the turbine
BSAP-POST / BSAP-DYNAM	Post-processing of BSAP program
BSAP-PRE	Pre-processing of BSAP program

APPENDIX 3.8.B

VISUAL INSPECTION ACCEPTANCE

CRITERIA FOR STRUCTURAL STEEL AND MISCELLANEOUS

STEEL WELDING TO MEET DESIGN REQUIREMENTS

3.8.B.1 Scope

This appendix provides the acceptance criteria for visual inspection of the welding of structural steel and miscellaneous steel. These criteria represent design requirements consistent with the engineering approval specified in AWS D1.1-75, Sections 1.1.2, 3.1.4, 3.7.4, 3.7.5 and AWS D1.1-85, Section 1.1.1.1. The criteria are applicable to structural systems subjected to static loading for which fracture resistance and fatigue resistance are not principal concerns. Where a question arises as to the classification of weld joints or the acceptance criteria, the Construction Manager shall be consulted for disposition.

The criteria of Section 3.8.B.3 were used prior to August 15, 1985 and that of Section 3.8.B.4 is used subsequently. Section 3.8.B.2 delineating classification of weld joints is applicable to the criteria of Section 3.8.B.3 only; it is not applicable to the final acceptance criteria of section 3.8.B.4. The criteria of Section 3.8.B.4 are the direct implementation for South Texas Project Electric Generating Station (STPEGS) of the Nuclear Construction Issues Group Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants (VWAC Revision 2) dated May 7, 1985.

In addition to the welding of structural and miscellaneous steel, Section 3.8.B.3 also pertained to welding of light gauge material in HVAC ductwork and other systems which are not specifically covered by AWS D1.1. The criteria of Section 3.8.B.4 do not pertain to welding of light gauge material in HVAC ductwork not specifically covered by AWS D1.1. Subsequent to August 15, 1985, the acceptance criteria for welding of light gauge material in HVAC ductwork are as set forth in STPEGS construction specifications.

3.8.B.2 Classification of Weld Joints (Applicable to Section 3.8.B.3 only)

The following classification of weld joints if determined by the intent of the engineering design and is based upon suitability for service requirements associated with each category.

3.8.B.2.1 <u>Category A Joints</u>. Category A Joints are part of the main building frame, including connections of the main building frame to embedded plates.

3.8.B.2.2 <u>Category B Joints</u>. Category B Joints are not part of the main building frame, but rather provide auxiliary support or framing for systems, components, and equipment. These joints are within the miscellaneous steel category, and shall include, but are not limited to, pipe supports (beyond the scope of ASME Codes), stairways, electrical tray and conduit supports, instrument supports, HVAC duct supports, heavy gauge HVAC ducts inside the RCB, and associated equipment. For connections of the foregoing items to the main building frame, to auxiliary steel and/or to embedded plates, when the connections consist of (1) gusset plates or flanges with continuous fillet welds on both sides of the plate element, or (2) members which are end-welded by all-round fillet welds to the attachment surface, the undercut criterion for Category A Joints shall apply for the whole connection. Typical cases of these connections subject to the Category A criterion for undercut are shown in Figure 3.8.B-2.

3.8.B.2.3 Category C Joints. Category C Joints are not part of the main building frame or auxiliary support system but rather perform a passive function. These joints may be within the miscellaneous steel category and may include, but are not limited to, doors, windows, hatch covers and frames, ledger, angles, handrails, kickplates, and grating.

3.8.B.2.4 Category D Joints. Category D Joints. Category D Joins are limited to those welds used in ductwork welding of thin-walled gauge steel which are not specifically covered by AWS D1.1.

3.8.B.3 Acceptance Criteria prior to August 15, 1985

Acceptance shall be based on the weld joint meeting each criterion listed for the applicable category. For welded connections between elements of two different joint categories, the integrity of the base metal of the elements of two different joint categories, the integrity of the base metal of the elements of the more stringent joint category shall be protected by verifying that welding undercut and/or arc strikes, if any, do not exceed the limits prescribed for the more stringent of the two joint categories. This provision is applicable only the base metal thicknesses of less than 3/8 in., unless otherwise noted on drawings or specifications.

3.8.B.3.1 Category A Joints.

3.8.B.3.1.1 Oversize Fillet Welds: The weld meets or exceeds specified size requirements. Either or both fillet weld legs may exceed design size by 1/8 in. for welds up to and including 5/16-in. fillet, and 1/4 in. for welds larger than 5/16-in. fillet.

Fillet welds exceeding the above limits may be considered acceptable if (1) the weld oversize is localized and cumulatively does not represent over 20 percent of the weld length or 2 in., whichever is longer, or (2) the fillet weld underwent prior repair that required deposition of additional weld metal.

Scalloping (intermittent melting of the plate edge) shall not be a cause for rejection of the weld as long as enough plate edge remains such that the fillet weld size can be verified.

Welds may be longer than specified. Continuous welds may be accepted in place of intermittent welds.

Welds may have end returns of nominal length equal to 2 times the weld size. Maximum length of return shall not exceed three times the weld size.

3.8.B.3.1.2 Undersize Fillet Welds: The fillet leg dimension may not underrun the nominal fillet size by more than 1/16 in., and the length of the underrun shall not be more than 10 percent of the weld length. For flange to web joints, the undersize may not be within two flange widths of the weld end.

Unequal leg fillet welds are acceptable, provided that the larger and smaller legs meet the prescribed oversize and undersize requirements.

3.8.B.3.1.3 Porosity: The weld may contain a maximum of 5 percent by surface area of unaligned, unclustered porosity. For aligned porosity, the sum of the diameters of piping porosity shall not exceed 3/8 inch in any linear inch of weld nor 3/4 inch in any 12-in. length of weld.

3.8.B.3.1.4 Weld Profile: Convexity height and butt weld reinforcement shall not exceed 1/8 in., except for welds 5/8 in. and over, where the convexity height or butt weld reinforcement shall not exceed 20 percent of weld size or thickness as long as the profile is smooth and free of sharp transitions.

3.8.B.3.1.5 Craters: The weld may have an underfilled crater, provided the underfill depth does not exceed 1/32 in. and the crater has a smooth contour blending gradually with the adjacent weld and base metal without acute notches.

3.8.B.3.1.6 Undercut: Undercut shall not exceed the value shown in Figure 3.8.B-1 for the Category A welds applicable to the area containing the undercut. Further, the undercut may be twice the value permitted by Figure 3.8.B-1 for an accumulated length of two inches in any 12 in. of weld, but in no case may undercut on one side be greater than 1/16 inch. For weld lengths less than 12 in., the permitted undercut length shall be proportional to the actual length.

3.8.B.3.1.7 Cracks: Cracks are unacceptable.

3.8.B.3.1.8 Fusion: Incomplete fusion between weld metal and base metal is unacceptable. Overlap is acceptable only if full fusion at the weld toe is visible.

3.8.B.3.1.9 Weld Spatter: Adherent weld spatter is acceptable unless its removal is required for further processing such as painting.

3.8.B.3.1.10 Arc Strikes: Every reasonable precaution shall be taken to prevent arc strikes due to welding or NDE work. If the arc strike is discovered during the welding operation, it shall be repaired at that time as required by AWS D1.1, Section 4.4. For high-strength low-alloy steels (minimum yield strength greater than 60,000 psi), all arc strikes shall be removed by grinding. The ground area shall be visually inspected to assure complete removal of the arc strike.

For other steels, if an arc strike is found at some subsequent time, it shall be visually examined and accepted if no cracking is evident. If cracking is evident, the repair shall conform with Section 4.4 of

AWS D1.1. In cleaning the arc strikes prior to visual examination, no power brushing or grinding shall be done.

3.8.B.3.2 Category B Joints.

3.8.B.3.2.1 Oversize Fillet Welds: Same as Section 3.8.B.3.1.1.

3.8.B.3.2.2 Undersize Fillet Welds: The fillet leg dimension may not underrun the nominal fillet size by more than 1/16 in., and the length of the underrun shall not be more than 20 percent of the weld length.

3.8.B.3.2.3 Porosity: Same as Section 3.8.B.3.1.3.

3.8.B.3.2.4 Weld Profile: Same as Section 3.8.B.3.1.4.

3.8.B.3.2.5 Craters: Underfilled groove weld craters are acceptable provided the depth of underfill is 1/16 in. or less. Underfill single-pass fillet weld craters are acceptable provided the crater length is less than 10 percent of the weld length. On multi-pass fillet welds a crater depth of 1/16 in. or less is acceptable.

3.8.B.3.2.6 Undercut: Undercut not exceeding 1/32-in. may be acceptable for the full length of the weld. Undercut not exceeding 1/16 in. may be accepted provided the width is greater than the depth and the undercut does not have an acute intersection at its root. The cumulative length of 1/16 in. undercut shall not exceed 50 percent of the weld length. For members welded from both sides, the cumulative undercut depth or length for both sides shall not exceed the above criteria applied to one side.

3.8.B.3.2.7 Cracks: Same as Section 3.8.B.3.1.7.

3.8.B.3.2.8 Fusion: Same as Section 3.8.B.3.1.8.

3.8.B.3.2.9 Weld Spatter: Same as Section 3.8.B.3.1.9.

3.8.B.3.2.10 Misalignment: Misalignment in butt welds not exceeding one-half the thickness of the thinner member thickness or 1/4 in., whichever is less, is acceptable.

3.8.B.3.2.11 Arc Strikes: Arc strikes are acceptable provided that the craters do not contain cracks as determined by visual examination. For high-strength low-alloy steels (minimum yield strength greater than 60,000 psi), all arc strikes shall be removed by grinding. The ground area shall be visually inspected to assure complete removal of the arc strike.

3.8.B.3.2.12 Backing Fit-up: The fit-up of a backing bar is not a basis for rejection.

3.8.B.3.3 Category C Joints.

3.8.B.3.3.1 Oversize Fillet Welds: Same as Section 3.8.B.3.1.1.

3.8.B.3.3.2 Undersize Fillet Welds: Same as Section 3.8.B.3.2.2.

3.8.B.3.3.3 Porosity and Slag Inclusions: Porosity and slag inclusions are not a basis for rejection.

3.8.B.3.3.4 Weld Profile: Convexity height and butt weld reinforcement shall not exceed 3/16 inch.

3.8.B.3.3.5 Craters: Same as Section 3.8.B.3.2.5.

3.8.B.3.3.6 Undercut: Undercut shall not exceed 3/32 in. or 25 percent of the material thickness, whichever is less.

3.8.B.3.3.7 Cracks: Same as Section 3.8.B.3.1.7.

3.8.B.3.3.8 Fusion: Same as Section 3.8.B.3.1.8.

3.8.B.3.3.9 Weld Spatter: Same as Section 3.8.B.3.1.9.

3.8.B.3.3.10 Misalignment: Same as Section 3.8.B.3.2.10.

3.8.B.3.3.11 Arc Strikes: Arc strikes are acceptable provided that the craters do not contain any cracks as determined by visual examination.

3.8.B.3.3.12 Backing Fit-up: Same as Section 3.8.B.3.2.12.

3.8.B.3.4 Category D Joints.

3.8.B.3.4.1 Oversize Fillet Welds: Same as Section 3.8.B.3.1.1.

3.8.B.3.4.2 Undersize Fillet Welds: Same as Section 3.8.B.3.2.2.

3.8.B.3.4.3 Porosity and Slag Inclusions: Porosity and slag inclusions are not a basis for rejection, provided the weld does not leak.

3.8.B.3.4.4 Weld Profile: Convexity height, overlap, and butt weld reinforcement may not exceed 1/8 inch. Profile of butt welds shall be convex. Faces of fillet welds may be slightly convex, flat, or slightly concave. Concavity shall not reduce the weld throat beyond that required for weld size.

3.8.B.3.4.5 Undercut: Undercut shall not exceed 50 percent of the material thickness.

3.8.B.3.4.6 Cracks: Same as Section 3.8.B.3.1.7.

3.8.B.3.4.7 Fusion: Incomplete fusion between weld metal and the base metal is unacceptable.

Butt welded joints for ductwork shall develop full penetration for a minimum of 80 percent of the length of the joint.

Corner welds used to seal ductwork are designated partial penetration welds. Such welds do not require full fusion, and weld reinforcement greater than the material thickness may constitute adequacy of the weld, provided the toes of the weld have complete penetration.

3.8.B.3.4.8 Weld Spatter: Same as Section 3.8.B.3.1.9.

3.8.B.3.4.9 Misalignment: Faying surfaces shall not exceed a 3/16 in. gap between parts to be joined. The leg of the fillet welds shall be increased by the amount of the separation.

Abutting parts to be joined by butt welds shall be carefully aligned. Misalignment shall not exceed the thickness of the thinner material being welded, as measured from the highest abutting member, nor more than 1/8 inch.

3.8.B.3.4.10 Arc Strikes and Scratching: Scratching of metal in fit-up and isolated arc strikes must be removed only to the extent necessary to remove sharp burrs. The intent of this stipulation is to preclude excessive grinding of the base metal, which shall not exceed 50 percent of the base metal thickness in the isolated areas.

3.8.B.3.4.11 Backing Fit-up: See Section 3.8.B.3.2.12.

3.8.B.3.4.12 Burn-through: Turning vanes and turning vane rails that are of light gauge material and welded to heavier gauge ductwork will be welded with a fillet weld as required by design drawings. Minor burn-through cannot be avoided on vanes and is permitted up to 1/4 inch in length, provided an equivalent length of fillet weld is added to compensate for the weld weakened by the burn-through.

Burn-through is permitted provided leak-tight integrity is maintained. Metal flow on the inside of the duct is permitted, provided it is fused completely with the parent metal and the metal thickness is not reduced by greater than 50 percent.

3.8.B.3.4.13 Distortion: Distortion caused by welding longitudinal seams shall not exceed 2 percent of the nominal diameter measured from the cross-sectional cord of the distorted area.

3.8.B.4 Acceptance Criteria after August 15, 1985

The following criteria are used for the acceptance inspection of welds in the uncoated condition. These criteria are also used for subsequent inspections after the welds have been coated, with the concurrence of the Engineer.

3.8.B.4.1 Weld Cracks. The weld shall have no cracks.

3.8.B.4.2 Fillets Weld Size. A fillet weld shall be permitted to be less than the size specified by 1/16 in. for 1/4 the length of the weld. Oversized fillet welds shall be acceptable if the oversized weld does not interfere with mating parts.

3.8.B.4.3 Incomplete Fusion. In fillet welds, incomplete fusion of 3/8 inch in any 4 in. segment, and 1/4 inch in welds less than 4 in. long, is acceptable. For groove welds, rounded end conditions that occur in welding (starts and stops) shall not be considered indications of incomplete fusion and are irrelevant.

3.8.B.4.4 Weld Overlap. Overlap is acceptable provided criteria for weld size and fusion can be satisfied. When fusion in the overlap length cannot be verified, an overlap length of 3/8 inch in any 4 in. segment, and 1/4 inch in welds less than 4 in. long, is acceptable.

3.8.B.4.5 Underfilled Craters. Underfilled craters shall be acceptable provided the criteria for weld size are met. Craters which occur outside the specified weld length are irrelevant provided there are no cracks.

3.8.B.4.6 Weld Profiles. The faces of fillet welds may be convex, flat, or concave, provided the criteria for weld size are met.

The faces of grove welds may be flat or convex.

Convexity of fillet and grove welds are not criteria for acceptance and need not be measured.

The thickness of groove welds is permitted to be a maximum of 1/32 in. less than the thinner member being joined.

3.8.B.4.7 Undercut. For material 3/8 in. and less nominal thickness, undercut depth of 1/32 in. on one side for the full length of the weld, or 1/32 in. on one side of for 1/2 the length of the weld and 1/16 in. for 1/4 the length of the weld on the same side of the member, is acceptable. For members welded on both sides where undercut exists in the same plane of a member, the cumulative lengths of undercut shall be limited to the lengths of undercut allowed on one side. Melt-through that results in a hole in the base metal is unacceptable.

For materials greater than 3/8 in. nominal thickness, undercut depth of 1/32 in. for the full length of the weld and 1/16 in. for 1/4 the length of the weld on both sides of the member is acceptable. When either welds or undercut exist only on one side of the member or are not in the same plane, the allowable undercut depth of 1/32 in. may be increased to 1/16 in. for the full length of the weld.

3.8.B.4.8 Surface Porosity. Only surface porosity whose major surface dimension exceeds 1/16 in. shall be considered relevant. Fillet and groove welds which contain surface porosity shall be considered unacceptable if:

- The sum of diamters of random porosity exceeds 3/8 inch in any linear inch of weld or 3/4 inch in any 12 in. of weld; or

- Four or more pores aligned and the pores are separated by 1/16 in. or less, edge to edge.

3.8.B.4.9 Weld Length And Location. The length and location of welds shall be as specified on the detail drawing, except that weld lengths may be longer than specified. For weld lengths less than 3 in., the permissible underlength is 1/8 in. and for welds 3 in. and longer the permissible underlength is 1/4 inch. Intermittent welds shall be spaced within 1 in. of the specified location.

For the outstanding legs of clip angles in framed beam connections, the fillet welds may have end returns of nominal length equal to 2X (weld size), but the maximum length of return shall not exceed 3X (weld size) unless otherwise shown in the design drawing.

3.8.B.4.10 Arc Strikes. Arc strikes and associated blemishes are acceptable provided no cracking is visually detected.

3.8.B.4.11 Surface Slag And Weld Spatter. Slag whose major surface dimension is 1/8 in. or less is irrelevant. Isolated surface slag that remains after weld cleaning and which does not exceed 1/4 in. in its major surface dimension, is acceptable. (Slag is considered to be isolated when it does not occur more frequently than once per weld or more than once in a 3-in. weld segment.) Spatter remaining after the cleaning operation is acceptable.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients. The following five operating conditions as defined in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code have been considered in the design of the mechanical systems and components.

3.9.1.1.1 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, is a normal condition.

3.9.1.1.2 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 is an upset condition. 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.

3.9.1.1.3 Emergency Conditions (Infrequent Incidents): Those deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system are emergency conditions. The conditions have a low probability of occurrence but are included to provided assurance that no gross loss of structural integrity results as a concomitant effect of any damage developed in the system.

3.9.1.1.4 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 consideration of public health and safety are involved are faulted conditions. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

3.9.1.1.5 Testing Conditions: Testing conditions are those pressure overload tests including hydrostatic tests, pneumatic tests, and leak tests specified. Other types of tests are classified under normal, upset, emergency, or faulted conditions.

To provide the necessary high degree of integrity for the equipment in the Reactor Coolant System (RCS), the transient conditions selected for equipment fatigue evaluation have been based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses have been based upon engineering judgement and experience.

The transients selected are representative of operating conditions which prudently should be considered to occur during plant operation and 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 have been given in the equipment specification for RCS components.

The design transients and the number of each that have been used for the current design basis fatigue evaluations are shown in Table 3.9-8. The number of transients stated in the descriptions below are the historical numbers used in the original design basis fatigue evaluations. In accordance with ASME B&PV Code, Section III, emergency and faulted conditions have not been included in fatigue evaluations.

3.9.1.1.6 Normal Conditions: The following primary system transients have been considered normal conditions:

1. Heatup and cooldown at 100°F per hour
2. Unit loading and unloading at 5 percent of full power per minute
3. Step load increase and decrease of 10 percent of full power
4. Large step load decrease with steam dump
5. Steady-state fluctuations
6. Feedwater (FW) cycling at hot shutdown
7. Loop out of service
8. Unit loading and unloading between 0 and 15 percent of full power
9. Boron concentration equalization
10. Refueling
11. Turbine roll test
12. Primary side leak test
13. Secondary side leak test
14. Tube leakage test

3.9.1.1.6.1 Heatup and Cooldown at 100°F Per Hour – The design heatup and cooldown cases have been conservatively represented by continuous operations performed at a uniform rate temperature change of 100°F per hour. (Administratively, heatup and cooldown rates are normally limited to values lower than the design rates to provide margin for short-term transients which can vary the actual rate.)

For these cases, the heatup occurs from the 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 the following.

1. Material ductility considerations which establish maximum permissible temperature rates of change, as a function of plant pressure and temperature, which are below the design rate of 100°F per hour.

2. Slower initial heatup rates when using pump energy only.

3. 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 5 such occurrences per year for the 40-year plant design life.

3.9.1.1.6.2 Unit Loading and Unloading at 5 Percent of Full Power Per Minute – The unit loading and unloading cases have been 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.

3.9.1.1.6.3 Step Load Increase and Decrease of 10 Percent of Full Power – The ±10 percent step change in load demand is a transient which has been 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 has been 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 of 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 average temperature is ultimately reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant temperature setpoint change is made as a function of turbine generator load as determined by first-stage turbine pressure measurement. The pressurizer pressure also decreases from its peak pressure value and follows 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 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 is raised to a value above its initial equilibrium value at the beginning of the transient.

The number of each operation is specified at 2,000 times or 50 per year for the 40-year plant design life.

3.9.1.1.6.4 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 (SG) safety valves. Thus, since these plants have been designed to accept a step decrease of 50 percent from full power, without reactor trip, the Steam Dump System provides the heat sink to accept 40 percent of the turbine load. The remaining 10 percent of the total step change is assumed by the Reactor Control System (control rods). If a steam dump system were not provided to cope with this transient, there would be such a strong mismatch between what the turbine is asking for and what the reactor is delivering that a reactor trip and lifting of SG valves would occur.

The number of occurrences of this transient is specified at 200 times or 5 per year for the 40-year plant design life.

3.9.1.1.6.5 Steady-State Fluctuations - It has been 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 have been considered:

1. Initial Fluctuations

   These are due to control rod cycling during the first 20 full-power months of reactor operation. Temperature is assumed to vary by ±3°F and pressure by ±25 psi, once during each 2-minute period. The total number of occurrences is limited to 1.5×10^5. These fluctuations have been assumed to occur consecutively, and not simultaneously with the random fluctuations.

2. 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 3.1×10^6.

3.9.1.1.6.6 Feedwater Cycling at Hot Shutdown - These transients can occur when the plant is at no-load condition, during which intermittent feeding of FW at 32°F into the SGs 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 2,000 times over the life of the plant.

3.9.1.1.6.7 Loop Out of Service – Operation with one loop out of service and the reactor critical is not permitted due to the impact on the Chapter 15 safety analysis. However, structural analysis has been performed that assumes the reactor power level is reduced and a single reactor coolant pump (RCP) is tripped.

The analysis assumes that this transient occurs twice per year or 80 times in the life of the plant. Conservatively, it has been assumed that all 80 occurrences can occur in the same loop. In other words, it has been 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 has been assumed that an inactive loop is inadvertently started up at maximum allowable power level ten times over the life of the plant. This transient is covered under “Upset Conditions.” Thus, the normal startup of an inactive loop has been assumed to occur 70 times during the life of the plant.

3.9.1.1.6.8 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 changes occur during unloading.

Prior to loading, it is assume that the plant is at hot shutdown condition, with cycling FW at 32°F. During the two-hour period following the beginning of loading, the FW temperature increases from 32°F to 300°F due to steam dump and turbine startup heat input to the FW. Subsequent to unloading, FW heating is terminated, steam dump is reduced to residual heat removal requirements, and FW 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.

3.9.1.1.6.9 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 heaters, thus causing a pressure increase, which will initiate spray at a compensated pressurizer pressure of approximately 2,275 psia. The proportional sprays return the pressure to 2,250 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.

3.9.1.1.6.10 Refueling – At the end of plant cooldown, the temperature of the fluid in the RCS is 140°F or less. 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 (RWST), which is conservatively assumed to be at 32°F, into the loops by means of the low-head safety injection (LHSI) pumps. It is conservatively assumed that the cold water is replaced with the colder water within ten minutes.

This operation is assumed to occur twice per year or 80 times over the life of the plant.

3.9.1.1.6.11 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. This transient occurs before plant startup and the number of cycles is independent of other operating transients.

3.9.1.1.6.12 Primary Side Leakage Test – Subsequent to each time the primary system is opened, a leakage test will be performed. During this test the primary system pressure is, for design purposes, raised to 2,500 psia, with the system temperature above the minimum temperature imposed by reactor vessel material ductility requirements, while the system is checked for leaks.

During this leakage test, the secondary side of the SG must be pressurized so that the pressure differential across the tube sheet does not exceed 1,600 psi. This is accomplished with the steam, FW, 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.

3.9.1.1.6.13 Secondary Side Leakage Test – During the life of the plant, it may be necessary to check the secondary side of the SG (particularly the manway closure) for leakage. For design purposes it is assumed that the SG 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 material ductility requirements. It is assumed that this test is performed 80 times during the 40-year life of the plant.

3.9.1.1.6.14 Tube Leakage Test – During the life of the plant, it may be necessary to check the SG for tube leakage and tube-to-tubesheet 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 SG is pressurized with water or water pressurized with gas, 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 may be repressurized to a higher pressure and the underside of the tube sheet is again checked for leaks. This process may be repeated until all the leaks are repaired or other NDE examinations of the tubes are performed to ensure that all leaking tubes 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 of Occurrences
200	400
400	200
600	120
840	80

Both the primary and secondary sides of the SGs will be at ambient temperatures during these tests.

3.9.1.1.7 Upset Conditions: The following primary system transients have been considered upset conditions:

1. Loss of load (without immediate reactor trip)
2. Loss of power
3. Partial loss of flow
4. Reactor trip from full power
5. Inadvertent RCS depressurization
6. Inadvertent startup of an inactive loop
7. Control rod drop
8. Inadvertent Emergency Core Cooling System (ECCS) actuation
9. Operating Basis Earthquake (OBE)
10. Excessive FW flow
11. RCS Cold Overpressurization

3.9.1.1.7.1 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 have been provided as part of the RPS, transients of this nature are not expected but have been 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.

3.9.1.1.7.2 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 RCPs are deenergized and, following coastdown of the RCP, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the SGs, which at this time are receiving FW, assumed to be at 32°F, from the Auxiliary Feedwater System (AFWS) operating from diesel generator power. Steam is removed for reactor cooldown through SG atmospheric safety and power operated relief valves provided for this purpose.

The number of occurrences of this transient is specified at 40 times or once per year for the 40-year plant design life.

3.9.1.1.7.3 Partial loss of Flow – This transient applies to a partial loss of flow from full power, in which an RCP 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 a turbine trip, on low reactor coolant flow, followed by automatic opening of the steam dump valves and flow reversal in the affected loop. The flow reversal causes reactor coolant at cold leg temperature to pass through the SG 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 sizeable 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.

3.9.1.1.7.4 Reactor Trip From Full Power – 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 SG. This is the result of continued heat transfer from the reactor coolant in the SG. The transient continues until the reactor coolant and SG secondary side temperatures are in equilibrium at zero-power conditions. A continued supply of FW and controlled dumping of steam remove the core residual heat and prevent the SG 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 transient 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 at 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:

1. Reactor trip with no inadvertent cooldown – 230 occurrences

2. Reactor trip with cooldown but no safety injection – 160 occurrences

3. Reactor trip with cooldown actuating safety injection – 10 occurrences

For design purposes, 20 occurrences of the reactor trip with no inadvertent cooldown (case a – 230 occurrences total) have been assumed to be accompanied by an emergency turbine overspeed. This situation could be caused by malfunction of the Turbine Control System following a large step load decrease with steam dump resulting in turbine speed increase past the turbine overspeed trip setpoint.

It has been assumed that the reactor trips and that the speed increases to 120 percent of nominal, with accompanying proportional increases in generator bus frequency, RCP speed, and reactor coolant flow rate.

3.9.1.1.7.5 Inadvertent Reactor Coolant System Depressurization – Several events can be postulated as occurring during normal plant operation which will cause rapid depressurization of the RCS. These include:

1. Actuation of a single pressurizer safety valve

2. Inadvertent opening of one pressurizer power-operated relief valve (PORV) due either to equipment malfunction or to operator error

3. Malfunction of a single pressurizer pressure controller, causing one PORV and two pressurizer spray valves to open

4. Inadvertent opening of one pressurizer spray valve, due either to equipment malfunction or to operator error

5. Inadvertent auxiliary spray

An inadvertent auxiliary spray actuation could occur if the auxiliary spray isolation valve is inadvertently opened during normal plant operation. This can only occur due to component failure or human error since the isolation valve does not receive an automatic actuation signal. This plant condition is conservatively categorized as an upset condition for RCS design purposes. The auxiliary spray piping design up to and including the pressurizer spray nozzle is analyzed as an emergency plant condition.

Of these events, the pressurizer safety valve actuation causes the most severe transients, and has been used as no “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 ECCS is actuated. Also, the passive accumulators of the ECCS are actuated when pressure decreases by approximately 1,600 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 is energized.

With pressure constant and safety injection (SI) 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. The number of inadvertent auxiliary spray cycles assumed to occur during plant life is 10.

3.9.1.1.7.6 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 RCP 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.

3.9.1.1.7.7 Control Rod Drop – This transient occurs if a bank of control rods drops into the fully inserted position due to a single component failure. The reactor is tripped on low pressurizer pressure. It is assumed that this transient occurs 80 times over the life of the plant.

3.9.1.1.7.8 Inadvertent Emergency Core Cooling System Actuation – A spurious SI signal results in an immediate reactor trip followed by actuation of the high-head safety injection (HHSI) and LHSI pumps. These pumps, however, do not deliver flow to the RCS, as both have shutoff heads below the minimum RCS pressure reached during the transient. This transient behaves similarly to the reactor trip from full power, with controlled steam dump and FW flow removing core residual heat after the trip. Reactor coolant temperature and pressure decrease as the control rods move into the core.

At the end of this transient, it is assumed that the plant is returned to noload condition, with pressure and temperature changes controlled within normal limits.

For design purpose, this transient has been assumed to occur 60 times during the 40-year design life of the plant.

3.9.1.1.7.9 Operating Basis Earthquake – The mechanical stresses resulting from the OBE have been considered on a component basis. Fatigue analysis, where required by the codes, has been performed by the supplier as part of the stress analysis report. The earthquake loads are 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 have been considered, however, in the design analysis. For the Nuclear Steam Supply System (NSSS) vendor scope of study, the number of occurrences for fatigue evaluation has been assumed to be 5 earthquakes at 10 cycles each (50 cycles total). For the balance-of-plant (BOP) scope of study, OBE is as defined in Section 3.7.3.2.2.

3.9.1.1.7.10 Excessive Feedwater Flow – This transient is defined only for the purpose of determining the adequacy of the SG, the RCS, and the pressurizer to withstand the effects of excessive FW flow. The pressure and temperature variations are considered in connection with analyzing the primary and secondary sides of the SG, the RCS, and the pressurizer.

This transient is conservatively defined as an umbrella case to cover occurrence of several events f the same general nature. These include:

1. Inadvertent opening of an FW control valve

2. Turbine overspeed (110 percent) with an open FW control valve

3. Small steam break with an open FW control valve

The excessive FW flow transient results from inadvertent opening of an FW control valve when the plant is at hot shutdown and the SG is in the no-load condition. The FW, Condensate, and Heater Drains Systems are in operation. The steam of an FW control valve has been assumed to fail, with the valve immediately reaching the full-open position. The FW flow to the affected loop is assumed to step from essentially zero flow to the value determined by the system resistance and the developed head of all operating FW pumps, with no main feedwater flow to the other loops. Steam flow is assumed to remain at zero, and the temperature of the FW entering the SG is conservatively assumed to be 32°F. FW flow is terminated by the SG high-high water level signal. After FW isolation, an eventual SG low-low water level signal initiates auxiliary feedwater (AFW). AFW flow is assumed to continue, with all pumps discharging into the affected SG. It is also assumed, for conservatism in the secondary side analysis, that AFW flows to the SGs not affected by the malfunctioned valve, in the so-called “unfailed loops”. Plant conditions stabilize at the values reached in 600 seconds, at which time AFW 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 AFWS under manual control.

For design purposes, this transient has been assumed to occur 30 times during the 40-year life of the plant.

3.9.1.1.7.11 RCS Cold Overpressurization: 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 usually generated by any one of a variety of malfunctions or operator errors. All events may be categorized as belonging to either events resulting in the addition of mass (mass input transient) or events resulting in the addition of heat (heat input transient). All of these possible transients are represented by composite “umbrella” design transients, referred to here as RCS cold overpressurization.

For design purposes, this transient has been assumed to occur 10 times during the 40-year design life of the plant.

3.9.1.1.8 Emergency Conditions: The following primary system transients have been considered emergency conditions:

1. Small Loss-of-Coolant Accident (LOCA)
2. Small steam line break
3. Complete loss of flow

3.9.1.1.8.1 Small Loss-of Coolant Accident – For design transient purposes, the small LOCA is defined as a break equivalent to the severance of a 1-in. inside diameter branch connection. (Breaks smaller than 0.375-in. inside diameter can be handled by the normal makeup system and produce no significant fluid systems transients.) Breaks which are much large than1 in. will cause accumulator 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 should be assumed that the ECCS is actuated immediately after the break occurs and subsequently delivers water at a minimum temperature of 32°F to the RCS.

3.9.1.1.8.2 Small Steam Line Break – For design transient purposes, a small steam line break has been 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:

1. The reactor is initially in a hot, zero-power condition.
2. The small steam line break results in immediate reactor trip and ECCS actuation.
3. A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.

4. The ECCS operates at a design capacity and repressurizes the RCS within a relatively short time.

3.9.1.1.8.3 Complete Loss of Flow – This accident involves a complete loss of flow from full power resulting from simultaneous loss of power to all RCPs. 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 40-year life of the plant.

3.9.1.1.9 Faulted Conditions: The following primary system transients have been considered faulted conditions. Each of the following accidents has been evaluated for one occurrence:

1. Reactor coolant loop (RCL) branch pipe break
2. Large steam line break
3. FW line break
4. RCP locked rotor
5. Control rod ejection
6. SG tube rupture
7. Safe Shutdown Earthquake (SSE)

The original design basis postulated pipe break locations in the RCL are described in Reference 3.6-1. The primary RCL components and support designs were based on these postulated break locations. A detailed fracture mechanics evaluation, as described in Reference 3.6-14, and 3.6-21 through 3.6-29, demonstrates that the probability of rupturing the RCL piping, pressurizer surge line, and the three SIS accumulator lines is extremely low under design basis conditions. Therefore, postulated ruptures in the RCL piping, pressurizer surge line, and the three SIS accumulator lines, and the following associated dynamic effects are not included in the design basis: missile generation, pipe whip, break reaction forces, jet impingement forces, decompression waves within the ruptured pipe, and pressurization in cavities, subcompartments and compartments.

3.9.1.1.9.1 Reactor Coolant Loop Branch Pipe Break – Following rupture of a RCL branch pipe resulting in a 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 ECCS is actuated to introduce water at a minimum temperature of 32°F into the RCS. The SI signal also results in reactor and turbine trips.

3.9.1.1.9.2 Large Steam Line Break – This transient is based on a complete severance of the largest steam line. The following conservative assumptions were made:

1. The reactor is initially in a hot, zero-power condition.

2. The large steam line break results in immediate reactor trip and in actuation of the SIS.

3.9.1.1.9.3 Feedwater Line Break – This accident involves a double-ended rupture of the main FW piping from full power, resulting in the rapid blowdown of one SG and the termination of main FW flow to the others. The blowdown is completed in approximately 43 seconds. Conditions were conservatively chosen to give the most severe primary side and secondary side transients. All AFW flow that is delivered to the faulted SG exits at the break. The incident is terminated when the operator manually terminates flow to the faulted loop.

3.9.1.1.9.4 Reactor Coolant Pump Locked Rotor – This accident is based on the instantaneous seizure of an RCP 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.

3.9.1.1.9.5 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 so that the pressurizer safety valves 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.

3.9.1.1.9.6 Steam Generator Tube Rupture – This accident postulates the double-ended rupture of an SG tube resulting in a decrease in pressurizer level and reactor coolant pressure. Reactor trip will occur due to the resulting SI signal. In addition, SI actuation automatically isolates the FW lines, by tripping all FW pumps and closing the FW isolation valves. When this accident occurs, some of the reactor coolant blows down into the affected SG, 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 SG. 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.

3.9.1.1.9.7 Safe Shutdown Earthquake – The mechanical dynamic or static equivalent loads due to the vibratory motion of the SSE have considered on a component basis.

3.9.1.1.10 Test Conditions: The following primary system transients under test conditions are discussed:

1. Primary side hydrostatic test

2. Secondary side hydrostatic test

3.9.1.1.10.1 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 has been performed at a water temperature which is compatible with reactor vessel material ductility requirements and a test pressure of 3,107 psig (1.25 times design pressure). In this test, the RCS has been pressurized to 3,107 psig coincident with SG 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 inservice inspection requirements of ASME Section XI. 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.

3.9.1.1.10.2 Secondary Side Hydrostatic Test – The secondary side of the SG 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 SG 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. The number of cycles is therefore independent of other operating transients.

3.9.1.2 Computer Programs Used in Analyses.

3.9.1.2.1 NSSS Systems and Components: For the NSSS scope of study, the following Westinghouse Electric Corporation-developed computer programs have been used in dynamic and static analyses to determine mechanical loads, stresses, and deformation of seismic Category I components and equipment. These are described and verified in References 3.9-1, 3.9-7 and 3.9-16.

1. WESTDYN – Static and dynamic analysis of redundant piping systems

2. FIXFM-3 – Time-history response of three-dimensional structures

3. WESDYN-2 – Piping system stress analysis from-time history

4. THRUST – Hydraulic loads on loops on loop components from blowdown information

5. WESAN – Reactor coolant loop equipment support structures analysis and evaluation

6. WECAN – Finite element structural analysis

7. ICES STRUDL-II – Linear elastic frame analysis of RCS support structure

8. MULTIFLEX – Thermal-hydraulic structure system dynamics

3.9.1.2.2 BOP System and Components: For the BOP scope of study, the following public domain and/or Bechtel Power Corporation-developed computer programs have been used.

3.9.1.2.2.1 ME101 Program – ME101 is a finite element computer program which performs linear elastic analysis of piping systems using standard beam theory techniques. The input data format is specifically designed for pipe stress engineering and the English system of units is used. Capabilities of ME210, ME909, ME912, ME913 softwares are included in the current version of ME101.

ME101 is used for the static and seismic analysis of the piping systems. Static analysis considers one or more of the following: thermal expansion, dead weight, uniformly distributed loads, and externally applied forces, moments, displacements, and rotations. Seismic analysis is based on standard normal mode techniques and uses response spectrum data. Two methods of eigenvalue solution are available. The Determinant Search or Subspace Iteration subroutines consider all data points as mass points. Kinematic Reduction and Householder QR Subroutines consider masses only at specified data points in designated directions.

Responses of the various modes are combined using the square root sum of the squares (SRSS) rule. Further, the responses of x-, y-, and z-earthquakes are combined using the SRSS rule. In a response spectrum seismic analysis, if some or all of the modes are closely spaced, ME101 combines the various modes based upon the grouping method per equation 4 of Regulatory Guide (RG) 1.92.

For verification, ME101 results have been compared against the following:

1. ME632, Computer Program, Seismic Analysis of Piping Systems, VERB MOD8, 1976 Bechtel Inernational Corporation, San Francisco, California

2. Pressure Vessel and Piping 1972 Computer Programs Verification, The American Society of Mechanical Engineers

3. Hand Calculations

3.9.1.2.2.2 SUPERPIPE – This program is used for static and dynamic piping analysis, both response spectrum and time-history types of loading. This is described and verified in Reference 3.9-14.

3.9.1.2.2.3 ADLPIPE – This program is used for Linear Elastic Analysis of Piping. Developed by Arthur D. Little, Inc. This is described and verified in Reference 3.9-15.

3.9.1.2.2.4 DYNAPO4 (NPS Piping Analysis Program) – This is described and verified in Reference 3.9-18.

3.9.1.2.2.5 CE798 – Structural Analysis System – The Structural Analysis System (ANSYS) computer program is a large-scale general purpose computer program. Analysis capabilities include static and dynamic; elastic, plastic, creep and swelling; small and large deflections; steady state and transient heat transfer and fluid flow. The program has been verified by comparison with known theoretical solutions, experimental results, and by other calculated solutions.

3.9.1.2.2.6 ME916 Program – The ME916 Program calculates nuclear Class 1 and 2 piping stresses at lug type integral attachments. The program has been verified based upon hand calculations.

3.9.1.2.2.7 BISEPS Program, Bechtel Interactive System for Engineering Pipe Support – The BISEPS program is an integrated computer system used in the design of pipe supports. This program is divided into two subprograms:

1. BISEPS STAND (ME140) performs the sizing of support hardwares, standard steel configurations, and welds in accordance with the standard support design specification. BISEPS STAND also generates a design "hard copy" or sketch showing the bill of materials, location plan and elevation, and other information used in the design, such as loads, movements, etc.

2. BISEP FRAME (ME240) performs only the sizing of nonstandard or skew steel configurations or any configuration not scoped within the BISEPS STAND program.

The program is verified by comparison with results from computer program BSAP described in Section 3.9.1.2.2.15.

3.9.1.2.2.8 ME913 Program – The ME913 program consists of numerical calculations of stress intensity levels for class 1 nuclear power piping components to validate their design adequacy.

The program determines the stress intensity levels of Class 1 nuclear power piping components for Equations 9 through 14 f Subarticle NB-3650, Analysis of Piping Components of Section III, ASME B&PV Code.

Prior to running this program, the user analyzes the piping systems using flexibility analysis program ME101 and heat transfer program ME643. The inputs to this program are the following:

1. Design pressure and temperature

2. Specified conditions

3. Design cycles

4. Piping configuration

5. Piping and piping component properties

6. Moment reactions due to:

    a. Thermal expansion loads

    b. Weight loads

    c. Earthquake loads

    d. Anchor movements

7. The thermal response of the piping system due to the specified transients: ΔT_1, ΔT_2, T_a and T_b values for the selected points in the system

The verification of ME913 is performed in two phases:

1. Phase I: Comparison of Results Between ASME Sample Problem (1) and ME913

   A comparison of stresses for ME913 and ASME sample problem is shown in Table 3.9-21. The results obtained from ME913 are different from those of the ASME sample problem but the difference is acceptable due to the high conservatism built into ME913. The higher stresses calculated by ME913 are due to the change of stress indices in the 1974 version as compared with the sample problem which adopts the 1971 version of ASME Section III.

2. Phase II: Hand Calculated Verification of the Computer Output

   The main FW piping system inside the Containment on the Grand Gulf project was analyzed using the ME913 program. A comparison of the tabulated stresses shown in Table 3.9-22 indicates almost identical results.

   The comparison between ME913 outputs and hand calculated results demonstrates the correct application of code equations. The slight numerical difference is mainly due to round-off errors in the desk calculator multiplications as compared with the numerical accuracy of the digital computer. Capabilities of this software are included in the current version of ME101.

3.9.1.2.2.9 ME643-1, ME643-2 and ME643-3 Program – The purpose of this program is to determine the temperature and stress distributions within a body as a function of time when subjected to thermal and/or mechanical loads. The program is valid for axisymmetric or plane structures and typically is used for gross or local discontinuity analysis as described in Paragraphs NB-3213.2 and NB-3213.3 of the ASME Code, Section III.

The program consists of three parts, each of which can be used separately. The first part, ME643-1, calculates steady state or transient temperature distributions due to temperature or heat flux inputs. The method used is the finite-element technique couple with a step-by-step time integration procedure. The program adopts a stepwise description of environmental temperatures and heat transfer coefficients if they are time dependent. Transient temperature distributions are calculated from the specified initial temperature and the step function heat inputs. ME643-1 is for plane and axisymmetrical structures.

The second part of the program, ME643-2, is built on the displacement method of the matrix theory of structures which calculates the displacements and stresses within the solids with orthotopic, temperature-dependent, nonlinear material properties. ME643-2 is also used for plane and axisymmetrical structures.

The third part of the program, ME643-3, calculates the steady state or transient temperature distribution due to temperature or heat flux inputs. The output of this program gives the code required parameters; i.e., Δt_t, Δt_2,, T_a, T_b, where Δt_1 is the linear thermal gradient, Δt_2 is the nonlinear thermal gradient, and T_a and T_b are the average temperature on side a and b of a gross discontinuity. ME643-3 is for straight pipe only.

The user has the option of saving the results for Part 1 on an external tape. After reviewing the printout, the user can specify the transient states for the stress evaluations; Part 2 then picks up the necessary information from the tape and performs the calculations.

The program was verified by comparing the results of ME643-1-2-3 program with the solution of an identical problem obtained by hand calculation. The results of these calculations agreed.

3.9.1.2.2.10 ME351 Pipe Rupture Analysis Program (PIPERUP) – The PIPERUP computer program performs nonlinear elastic-plastic analysis of three-dimensional piping systems subjected to concentrated static or dynamic time history forcing functions. The program solutions have been verified by test results, closed from solution and/or independent computer programs.

3.9.1.2.2.11 ME210 Program, Local Stress in Cylindrical Shells Due to External Loading – The ME210 computes the local stresses in cylindrical shells that result from external loadings. The program is based on Welding Research Council Bulletin 107, August 1965. The program has been verified based upon hand calculations. Capabilities of this software are included in the current version of ME101.

3.9.1.2.2.12 ME909 Spectra Curves Merging – This program merges response spectra curves, makes a neutral plot file of these curves, and produces data cards from ME101 seismic analysis. The program is verified by hand calculations. Capabilities of this software are included in the current version of ME101.

3.9.1.2.2.13 ME912 Pipe Thermal Transient Program – This program is a quasi-two-dimensional transient analysis program. It computes radial thermal gradients through pipe walls, reduced thermal transients in the axial direction and axial discontinuity temperature differences. The program was verified by comparing the results of ME-912 with the solution of an identical problem obtained by hand calculation. Capabilities of this software are included in the current version of ME101.

3.9.1.2.2.14 ME035 Baseplate – This finite element program is used for design and/or analyzing of baseplate for pipe supports. Baseplate has been verified against the results of the CE135 Baseplate II program (Section 3.9.1.2.2.21).

3.9.1.2.2.15 ME150 Frame Analysis Program for Pipe Supports (FAPPS) – This finite element program is used for analysis and design of pipe support frames, welds, and baseplate selections. FAPPS has been verified against CE901 STRUDL and manual calculations.

3.9.1.2.2.16 CE050-Bolts – This program determines concrete expansion anchor loads and interaction values for baseplates employing expansion anchors. The analysis considers the effects of plate flexibility, bolt stiffness, and attachment size. The program is verified by hand calculation.

3.9.1.2.2.17 CE901-ICES-STRUDEL (STRUDL) – A description of the program is provided in Appendix 3.8.A. Documentation of program verification is maintained in the Bechtel Information Services Central Library.

3.9.1.2.2.18 CE800 Bechtel Structural Analysis Program (BSAP) – A description of the program is provided is provided in Appendix 3.8.A. Documentation of program verification is maintained in the Bechtel Information Services Central Library.

3.9.1.2.2.19 CE201-BSAS-POST – A description of the program is provided in Appendix 3.8.A. Documentation of program verification is maintained in the Bechtel Information Services Library.

3.9.1.2.2.20 CE212-BSAP-PRE – A description of the program is provided in Appendix 3.8.A. Documentation of the program is maintained in the Bechtel Information Services Central Library.

3.9.1.2.2.21 CE035-BASEPLATE II – A description of the program is provided in Appendix 3.8.A. Documentation of the verification is maintained in the Bechtel Information Services Central Library.

3.9.1.2.2.22 CE413-WELD – The WELD program is used to size fillet welds for connections of wide flanges, tubes, pipes, angles, and channel. The program computes weld sizes based on AISC, NF, B31.1 and minimum weld for minimum heat transfer. The program is verified by hand calculations.

3.9.1.2.2.23 NE458-RELAP5 (MOD1 and MOD3)/NE565-REPIPE/NE457-R5FORCE - Thermal-Hydraulic Transient Analysis - RELAP5 and force post-processors REPIPE or R5FORCE are used for analysis of fluid transients in piping systems. RELAP5/MOD1 is used with the REPIPE post-processor and RELAP5/MOD3 is used with the R5FORCE post-processor. RELAP5 solves equations of conservation of mass, energy, and momentum in one dimension for steam and/or water flow. The effects of noncondensible gas on steam/liquid flow are considered in the equations. REPIPE and its successor program R5FORCE are post-processor programs that convert RELAP5 output control volume and junction results into pipe reaction forces in a format suitable for input to ME101. Documentation for the programs, including verification reports, is maintained by the Bechtel Information Services Central Library.

3.9.1.2.2.24 ME150 FAPPS – Frame Analysis Program for Pipe Supports (FAPPS) is an interactive computer program for the analysis and design of pipe supports. It optimizes member sizes, welds, baseplates and embedments based on various user–specified design limitations. The program allows load combination by algebraic, absolute, or SRSS methods. The program has been verified against Bechtel Standard Structural Analysis Program CE901 (STRUDL) and hand calculations.

3.9.1.2.2.25 ME035 BASEPLATE – ME035 is a finite element-computer program for the analysis and design of baseplate. The program has important features like automatic mesh generation, availability of standard attachments, multiple plate thicknesses, and different printout options. The program has been verified against CDC Baseplate II (Bechtel CE035).

3.9.1.2.2.26 ME225 ANCHORPLATE – ME225 is used to analyze and design interface anchors between nonseismic piping and seismically designed piping. This program ahs been verified by manual calculations.

3.9.1.2.2.27 ME602 Spectra Merging and Simplified Seismic Analysis. This program merges response spectra curves, makes a neutral plot of these curves, and produces data files for ME101 Seismic Analysis. In addition the program calculates seismic span. Documentation of program verification is maintained in the Bechtel Information Services Central Library. Capabilities of this software are included in the current version of ME101.

3.9.1.2.2.28 ME152 Standard Frame Analysis Program for Pipe Supports – (SMAPPS) is a user-friendly, interactive computer program for the analysis and design of pipe supports. It has complete analysis capabilities for six commonly encountered support frame configurations. Any pipe support frame that utilizes one of these six configurations can be completely analyzed to satisfy all design requirements for member stress, deflection, stiffness, welds and baseplates, all within one run. Documentation of program verification is maintained in the Bechtel Information Services Library.

3.9.1.2.2.29 ME226 Pipe Clamp – Pipe Clamp (PICLAMP) will design the components of six special cases of pipe support clamps. The program computes the minimum required thickness at two critical sections of the clamp. The program will also compute the stress in the clamp studs. The stresses in the stanchion and baseplate are computed when applicable, and the minimum weld size based on stress is computed. Finally, the program computes certain clamp dimensions and, based on the input data, computes the total weight of the clamp and its associated hardware. The program is verified by hand calculation.

3.9.1.2.2.30 PS+CAEPIPE – PS+CAEPIPE is a group of interrelated computer programs for performing linear elastic analysis of three-dimensional piping systems subject to variety of loading conditions. PIPESTRESS software is included in every copy of PS+CAEPIPE. PIPESTRESS has advanced static and dynamic analysis capabilities including detailed uniform and multi-level response spectrum analyses, time history and fatigue calculations, and multiple load cases and combinations. PIPESTRESS is nuclear quality assured on multiple computer platforms to obtain quality assured final reports. This is described and verified in Reference 3.9-20.

3.9.1.2.2.31 WESPLAT – WESPLAT is a non-linear finite element analysis code for the analysis of rectangular plates attached by anchors to a rigid foundation. WESPLAT utilizes a non-linear plate bending, finite element approach. The baseplate is modeled using an assembly of plate elements at nodal points. The plate is divided into elements dependent upon the user-specified problem. Reference 3.9-21 describes the use of WESPLAT.

3.9.1.2.2.32 NEWKFAC – This computer code is used to calculate pressure drops, resistances, and volumes of piping networks based on pipe size, type/number of fittings and flowrate in the system.

3.9.1.3 Experimental Stress Analysis. Experimental stress analysis methods have not been used for any seismic Category I ASME B&PV Code, Section III mechanical system or equipment.

3.9.1.4 Considerations for the Evaluation of the Faulted Conditions.

3.9.1.4.1 Stress Criteria for Class 1 Components and Supports in Nuclear Steam Supply System Scope: The structural stress analyses performed on NSSS components and component supports consider the loadings shown in Table 3.9-2.1. These loads result from thermal expansion, pressure, weight, OBE, SSE, RCL branch pipe break, and plant operational thermal and pressure transients.

3.9.1.4.2 Analysis of the Reactor Coolant Loop and Supports: The loads used in the analysis of the RCL piping are described in detail below:

1. Pressure

   Pressure loading has been 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 B&PV Code.

   The term "operating pressure' has been 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 RCL model at change in direction or flow area.

2. Weight

   A deadweight analysis has been performed to meet code requirements by applying a 1.0g 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 RCL seismic piping analyses have been derived from dynamic response analyses of the Reactor Containment Building (RCB) subjected to seismic ground motion. Input is in the form of floor response spectrum curves at various elevations within the RCB.

   For the OBE and SSE seismic analyses, 2 and 4 percent critical damping, respectively, have been used in the RCL/supports system analysis.

   In the response spectrum method of analysis, the total response loading obtained from the seismic analysis consists of two parts: the inertia response loading of the piping system and the differential anchor movements loading. Two sets of seismic moments are required to perform an ASME Code analysis. The first set includes only the moments resulting from inertia effects, and these moments are used in the resultant moment, M_I, value for equations 9 and 13 of NB-3650. The second set includes the moments resulting from seismic anchor motions and is used in equations 10 and 11 of NO-3650. Differential anchor movement is discussed in Section 3.7.

4. Loss-of-Coolant Accident

Blowdown loads have been developed in the RCLs as a result of transient flow and pressure fluctuations following a postulated RCL branch pipe break. Structural consideration of dynamic effects of postulated pipe break requires postulation of a finite number of break locations. Postulated pipe break locations are given in Section 3.6.

Time-history dynamic analyses have been performed for these postulated break cases. Hydraulic models have been used to generate time-dependent hydraulic forcing functions used in the analysis of the RCL for each break case. For a further description of the hydraulic forcing functions, refer to Section 3.6.

5. Transients

The ASME B&PV Code requires satisfaction of certain requirements relative to operating transient conditions. Operating transients are tabulated in Section 3.9.1.1.

The vertical thermal growth of the reactor pressure vessel (RPV) nozzle centerlines has been considered in the thermal analysis to account for equipment nozzle displacements as an external movement.

The hot moduli of elasticity, E, the coefficient of thermal expansion at the metal temperature. α, 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 RCS, 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 operation conditions.

3.9.1.4.3 Reactor Coolant Loop 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, the response spectra method for seismic dynamic analysis, and time-history integration method for the LOCA dynamic analysis.

The integrated RCL/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 RCL piping and components, the stiffness of supports, and the stiffness of auxiliary line piping which affects 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 RCL/supports system model, constructed for the WESTDYN computer program, is represented by an ordered set of data which numerically describes the physical system. Figure 3.9-6 shows an isometric line schematic of this mathematical model. The SG and RCP vertical lateral support members are described in Section 5.4.14.

The spatial geometric description of the RCL model is based upon the RCL piping layout and equipment drawings. The node point coordinates and incremental length 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, α, the average temperature change from ambient temperature, ΔT, and the weight per unit length, are specified for each element. The primary equipment supports have been represented by stiffness matrices which define restraint characteristics of the supports. Due to the symmetry of the static loadings, the RPV centerline has been represented by a fixed boundary in the system mathematical model. The vertical thermal growth of the RPV nozzle centerline has been considered in the construction of the model.

The model is made up a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section have been used to define the stiffness matrix for the section. Using the transfer relationship for a section, the load required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section have been obtained. These loads have been 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 are 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 deadweight, thermal, and general pressure loading conditions are obtained by using the WESTDYN computer program. The derivation of the hydraulic loads for the LOCA analysis of the loop is covered in Section 3.6.2.

2. Seismic

The model used in the static analysis has been modified for the dynamic analysis by including the mass characteristics of the piping and equipment. The effect of the equipment motion on the RCL/supports system is obtained by modeling the mass and the stiffness characteristics of the equipment in the overall system model.

The SG has been typically represented by four discrete masses. The lower mass is located at the elevation of the lower support attachment point. The second mass has been located at the SG upper support elevation, the third mass has been located at the center of the upper shell, and the fourth mass is located at the top of the SG. The RCP has been 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 RPV and core internals have been typically represented by approximately ten discrete masses. The masses are lumped at various locations along the length of the vessel and along the length of the representation of the core internals.

The component lateral supports are inactive during plant heatup and cooldown and normal plant operating conditions. However, these restraints become active when the plant is at power and under the rapid motions of the RCL 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 have been performed at the full-power condition.

The response spectra method employs the lumped-mass technique, linear elastic properties, and the principle of modal superposition. The floor response spectra have been applied along both horizontal axes and the vertical axis simultaneously.

From the mathematical description of the system, the overall stiffness matrix, K, has been developed from the individual element stiffness matrices using the transfer matrix method. After deleting the rows and columns representing rigid restraints, the stiffness matrix has been revised to obtain a reduced stiffness matrix, K_R associated with mass degrees of freedom, only. From the mass matrix and the reduced stiffness matrix, the natural frequencies and the normal modes are determine.

The modal participation factor matrix is computed and combined with the appropriate response spectra value to give the modal amplitude for each mode. The total modal amplitude has been obtained by taking the SRSS of the contributions for each direction.

The modal amplitudes are then converted to displacements in the global coordinate system and applied to the corresponding mass point. From these data, the forces, moments, deflections, rotations, support reactions, and piping stresses have been calculated for all significant modes.

The total seismic response is computed by combing the contributions of the significant modes as described in Section 3.7.

3. Loss-of-Coolant Accident

The mathematical model used in the static analyses has been modified for the LOCA analyses by including the mass characteristics of the RCL piping and primary equipment. The natural frequencies and eigenvectors are determined from this model.

The time-history hydraulic forces at the node points have been combined to obtain the forces and moments acting at the corresponding structural limped-mass node points.

The dynamic structural solution for the full-power LOCA and steam line break has been 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 have been 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 has been performed in subprogram FIXFM-3. 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 RCL dynamic model have been determined with the WETDYN 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 has been obtained using subprogram FIXFM-3 and employing 4 percent critical damping.

The LOCA displacements of the RPV have been applied in time-history form as input to the dynamic analysis f the RCL. The LOCA analysis of the RPV includes all the forces acting on the vessel, including internals reactions, and loop mechanical loads. The RPV analysis is described in Section 3.9.1.4.6.

The time-history displacement response of the loop is used in computing support loads and in performing the stress evaluation of the RCL piping.

The support loads have been computed by multiplying the support stiffness matrix and the displacement vector at the support point. The support loads are used in the evaluation of the supports.

The time-history displacements of the FIXFM-3 subprogram have been used as input to the WETDYN-2 subprogram 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 RCL masses. The results of this solution have been used in the piping stress evaluation.

The resultant asymmetric external pressure loads on the RCP and steam generator resulting from a postulated pipe rupture and pressure buildup in the loop compartments are applied dynamically to the RCL model. This model is the same integrated RCL/supports system model used to compute loadings on the components, component supports and Reactor Containment (RC) 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. The resultant 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 RCL branch pipe breaks, as discussed in Section 3.6. The RCL piping is evaluated in accordance with the faulted condition criteria of ASME III, NB-3650 and appendix F. The loads included in the evaluation result from the SSE inertia loading, deadweight, pressure, RCL branch pipe break hydraulic forces, asymmetric subcompartment pressurization forces, jet impingement loads from postulated pipe breaks and reactor vessel motion. Individual loadings at critical stress locations are combined and primary stress intensities at all locations are within the faulted condition stress limit.

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 ASME B&PV Code into three parts, a uniform, a linear, and a nonlinear part. The uniform part results in general expansion loads; the linear part causes a bending moment across the wall; and the nonlinear part causes a skin stress.

The transients as defined in Section 3.9.1.1 are used to define the fluctuations in plant parameters. A one-dimensional finite difference heat conduction program has been used to solve the thermal transient problem. The pipe has been represented by about 100 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 on Figure 3.9-8.

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 the 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:

$$M = E\int_O^H \left(X - \frac{H}{2}\right) T(X,t)dx$$

The equivalent thermal moment produced by the linear thermal gradient as shown on Figure 3.9-8 about the mid-wall thickness is equal to:

$$M_L = E\alpha \frac{\Delta T_1}{12} H^2$$

Equating M_L and M, the solution for ΔT_1 as a function of time is:

$$\Delta T_1(t) = \frac{12}{H^2}\int_O^H (X - \frac{H}{2})\ T(X,t)dx$$

The maximum nonlinear thermal gradient, ΔT_2, occurs 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.

$$\Delta T_{2I}(t) = T(O,t) - T_A(t) - \frac{\Delta T_1(t)}{2}$$

5. Load Set Generation

A load set is defined as a set of pressure loads, moment loads, through-wall thermal effects, and the axial thermal gradient at a given location and time in each transient. The method of load set generation is based on Reference 3.92. 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. The axial thermal gradient, defined by 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.

- ΔT_1
- ΔT_2
- $\alpha_A T_A - \alpha_B T_B$
- Moment loads due to T_A
- Pressure loads

This procedure produces at least twice as many load sets as transients for each point.

As a result of the normal mode spectral technique employed in the seismic analysis, the load components cannot be given signed value. Eight load sets are used to represent all possible sign permutations of the seismic moments at each point, thus assuring that the most conservative combination of seismic loads is used in the stress evaluation.

For all possible load set combinations, the primary-plus-secondary and peak stress intensities, fatigue reduction factors, K_e, and cumulative usage factors, U, have been calculated. The WESTDYN program has been used to perform this analysis in accordance with the ASME B&PV Code, Section III, Subsection NB-3650. Since it is impossible to predict the order of occurrence of the transients over a 40-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 has been 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.

3.9.1.4.4 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: (1) to quantitatively represent the elastic restraints which the supports impose upon the lop, and (2) to 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 Section 5.4.14. Detailed models have been developed using beam elements and plate elements, where applicable.

The SG lower support is shown in Figure 3.9.13. The struts are represented by single-acting springs in the RCL analysis; the columns are modeled as individual double-acting springs. The SG upper support is shown in Figure 3.9-14. A model for the STRUDL (Ref. 3.9-1) computer program (figure 3.9-15) is constructed for the SG upper lateral support ring girder. Structure geometry, topology, member releases, and concrete flexibilities are included to accurately represent the behavior of the support system. Rigid spokes, extending from a point on the SG vertical axis to points where loads are transferred to the ring girder, are included in the model. The SG upper support model is used to determine the spring constants used to represent the support in the RCL model.

The RCP supports are shown in Figure 3.9-16. Single-acting springs represent the tie bars and double-acting springs represent the columns in the RCL model. The brackets of the compression and tension tie bars have slotted pin holes which make the members single-acting only.

A three-dimensional finite element model is used for the RPV support structure. The WECAN (Ref. 3.9-16) computer program is used for the support analysis.

For each operating condition, the loads (obtained from the RCL analysis) acting on the support structures are appropriately combined. The adequacy of each member of the SG supports and RCP supports is verified by solving the ASME III Subsection NF stress and interaction equations by means of hand calculations and the WECAN (Ref. 3.9-1) computer program. 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.

3.9.1.4.5 Analysis of Primary Components: Equipment which serves as part of the pressure boundary in the RCP loop includes the SGs, the RCP, the pressurizer, and the reactor vessel. This equipment is American Nuclear Society (ANS) Safety Class 1, and the pressure boundary meets the requirements of the ASME B&PV Code, Section III, Subsection NB. This equipment is evaluated for the loading combinations outlined in Table 3.9-2.1. The equipment is analyzed for: (1) the normal loads of deadweight, pressure, and thermal; (2) mechanical transients of OBE, SSE, and pipe ruptures, including the effects of asymmetric subcompartment pressurization; and (3) pressure and temperature transients outlined in Section 3.9.1.1.

The results of the RCL analysis have been used to determine the loads acting on the nozzles and the support/component interface locations. These loads have been supplied for all loading conditions on an “umbrella” loads basis; that is, on the basis of previous plant analyses, a set of loads has been 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 has been 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 have been handled by individualized analysis.

Seismic analyses have been performed individually for the RCP, the pressurizer, and the SG. Detailed and complex dynamic models have been used for the dynamic analyses. The response spectra corresponding to the building elevation at the highest component/building attachment elevation have been used for the component analysis. Seismic analyses for the SG have been performed using 2 percent damping for the OBE and 4 percent damping for the SSE. The analysis of the RCP for determination of loads on the motor, main flange, and pump internals has been 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 RCP since the main flange, motor stand, and motor are all bolted assemblies (Section 5.4). The RPV has been qualified by static stress analysis based on loads that have been derived from dynamic analysis.

Reactor coolant pressure boundary (RCPB) components have been further qualified to ensure against unstable crack growth under faulted conditions by performing detailed fracture analysis of critical areas of this boundary. Actuation of the ECCS produces relatively high thermal stresses in the system. Regions of the RCPB which come into contact with ECCS water are given primary consideration. These include the reactor vessel belt line region and the reactor vessel inlet nozzles.

The thermal effects in the regions of interest have been evaluated using fracture instability analysis and fatigue crack growth analysis. These analyses are based on linear elastic fracture mechanics (LEFM). The LEFM approach to the design against failure is basically a stress intensity consideration in which criteria are established for fracture instability in the presence of a crack. Consequently, a basic assumption employed in LEFM is that a crack or crack line defect exists in the structure. The essence of the approach is to relate the stress field developed in the vicinity of the crack tip to the applied stress on the structure, the material properties, and the size of defect necessary to cause failure.

The elastic stress field at the crack tip in any cracked body can be described by a single parameter designated as the stress intensity factor, K. The magnitude of K is a function of the geometry of the body containing the crack, the size and location of the crack, and the magnitude and distribution of the stress.

The criterion for failure in the presence of a crack is that failure will occur whenever the stress intensity factor exceeds some critical value. For the opening mode of loading (stresses perpendicular to the major plane of the crack), the stress intensity factor is designed as K_I and the critical stress intensity factor is designated K_{IC}. Commonly called the fracture toughness, K_{IC} is an inherent material property which is a function of temperature and strain rate. Any combination of applied load, structural configuration, crack geometry, and size which yields a stress intensity factor K_{IC} for the material results in crack instability.

The LEFM Analysis Methods in ASME XI, Appendix A and ASME III, Appendix G are used to perform the fracture evaluation of postulated flaws to establish that the vessel integrity is maintained.

In addition, it has been well established that the crack propagation of existing flaws in a structure subjected to cyclic loading can be defined in terms of fracture mechanics parameters. Thus, the principles of LEFM are also applicable to fatigue growth of a postulated flaw.

An example of a faulted condition evaluation carried out according to the procedure discussed above is given in Reference 3.9-3. This report discusses the evaluation procedure in detail as applied to a severe faulted condition (postulated LOCA) and concludes that the integrity of the RCPB would be maintained in the event of such an accident.

The pressure boundary portion of RCS Class 1 valves has been designed and analyzed according to the requirements of ASME B&PV Code, Section III, NB-3500.

Valves in sample lines connected to the RCS are not considered ANS Safety Class 1 nor ASME Class 1. This is because the nozzles where the line connects to the primary system piping are orificed to a 15/64-in. hole. This hole restricts the flow so that loss due to severance of one of these lines can be made up by normal charging flow.

3.9.1.4.6 Dynamic Analysis of Reactor Pressure Vessel for Postulated Loss of Coolant Accident:

3.9.1.4.6.1 Introduction – This section presents the method of computing the reactor pressure vessel response to a postulated RCL branch pipe break. The structural analysis considers simultaneous application of the time-history loads on the reactor vessel resulting from the RCL 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 RCLs with the primary supports of the steam generators and the RCPs.

3.9.1.4.6.2 Interface Information – All input information was developed within Westinghouse. This information includes reactor internals properties, loop mechanical loads and loop stiffness, internal hydraulic pressure transients, and reactor support stiffnesses. These inputs allowed formulation of the mathematical models and performance of the analyses, as will be described.

3.9.1.4.6.3 Loading Conditions – Following a postulated pipe rupture, the reactor vessel is excited by time-history forces. As previously mentioned, these forces are the combined effect of reactor coolant loop mechanical loads and reactor internal hydraulic forces.

The RCL mechanical forces are derived from the elastic analysis of the loop piping for the postulated break. This analysis is described in Section 3.9.1.4.3. The reactions on the nozzles of the RCL piping are applied to the vessel in the RPV blowdown analysis.

The reactor internal hydraulic pressure transients were calculated with the assumption that the structural motion is coupled with the pressure transients. This phenomena 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 (Ref. 3.9-7).

3.9.1.4.6.4 Reactor Vessel and Internals Modeling – The reactor vessel is restrained by two mechanisms: (1) the four attached reactor coolant loops with the SG and RCP 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 Section 5.4.14 and are shown in Figures 5.4-12 and 3.8.3-1. The support shoe provides restraint in the horizontal directions for reactor vessel motion.

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 (Figure 3.9-11) represents the reactor vessel shell and associated components. The reactor vessel is restrained by the four reactor vessel supports and by the attached primary coolant piping. Each reactor vessel support is modeled by a linear horizontal stiffness and a vertical nonlinear element with lift-off capability. The attached piping is represented by a stiffness matrix.

The second submodel (Figure 3.9-12) represents the reactor core barrel, neutron panels, lower support plate, 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 impact is represented by nonlinear elements at the core barrel flange, core barrel nozzle, and lower radial support location.

The third and innermost submodel (Figure 3.9-12A) represents the lower core support plate, guide tubes, support columns, upper core plate, and fuel. The third submodel is connected to the first and second by stiffness matrices and nonlinear elements.

3.9.1.4.6.5 Analytical Methods – The time-history effects of 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 intergrating 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 analysis. The resulting loads and stresses in the piping components and supports include both RCL branch pipe loop blowdown loads and reactor vessel displacements. Thus, the effect of vessel displacements upon loop response and the effect of RCL branch pipe blowdown upon vessel displacement are both evaluated.

3.9.1.4.6.6 Results of the Analysis – As described, the reactor vessel and internals were analyzed for postulated RCL branch pipe break locations. The maximum loads induced in the vessel supports due to the postulated pipe break are 2,135 kips vertical load (including deadweight) and 895 kips horizontal load. These loads are per vessel support and are applied at the vessel nozzle pad. It is conservatively assumed that the maximum horizontal and vertical loads occur simultaneously and on the same support, even though the time-history results show that these loads do not occur simultaneously on the same support.

3.9.1.4.7 Stress Criteria for Class 1 Components and Component Supports for Balance of plant Scope of Supply: All Class 1 components and supports have been designed and analyzed for the design, normal, upset, emergency and faulted conditions as specified in the rules and requirements of the ASME B&PV Code, Section III. Stress criteria for Class 1 BOP valves and piping are outlined in Tables 3.9-5 and 3.9-7. Stress limits for Class 1 BOP component supports are given in Table 3.9-7B.

The Class 1 piping has been designed and analyzed for the design, normal, upset, emergency and faulted conditions in accordance with the requirements of NB-3600 of the ASME B&PV Code, Section III, 1974 Edition through Winter Addenda of 1975, NB-3658 of Summer Addenda of 1977, NB-39650 and NB-3680 of Summer Addenda of 1979 and Table NB-3681(a)-1 of 1983 Edition of ASME Section III. When the stresses as determined by the methods given in NB-3630 exceed the limits thereof, the design can be accepted provided it meets the requirements of NB-3200. The rules of NB-3630 meet all the requirements of NB-3200.

3.9.1.4.8 Evaluation of the Control Rod Drive Mechanisms: The Control Rod Dive Mechanisms (CRDMs) are evaluated for the effects of postulated RCL branch pipe breaks. A time-history analysis of the CRDMs is performed for the vessel motion discussed in Section 3.9.1.4.6. A model of the CRDMs is formulated with gaps at the upper CRDM support 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. The combined effect including seismic loads is shown to be less than the allowable loads at all locations.

3.9.2 Dynamic Testing Analysis

3.9.2.1 Preoperational Vibration and Dynamic Effects Testing on Piping. Piping vibration tests were performed during the initial test program to comply with the recommendations of RG. 1.68 and satisfy the requirements of ASME B&PV Code, Section III. The following systems were visually inspected and measured (as needed) during the preoperational test programs:

- Auxiliary Feedwater
- Component Cooling
- Containment Spray (except spray header)
- Chemical and Volume Control
- Main Feedwater (safety-related portion only)
- Main Steam (safety-related portion only)
- Residual Heat Removal System
- Safety Injection System
- Essential Cooling Water System
- Diesel Generator
- Reactor Coolant
- Essential Chilled Water (safety-related portion only)
- Fuel Pool Cooling and Cleanup
- Reactor Coolant Pressurizer System (PORV Discharge Lines)
- Steam Generator Blowdown

Small bore piping was also included in the preoperational test program. Inspection of piping systems included both large bore and small bore piping. Additionally, essential safety-related instrumentation lines up to the first rigid guide support were included in the vibration monitoring program during preoperational testing. If observations suggested that other spans were being excited, further inspection was conducted on a case-by-case basis.

3.9.2.1.1 Nuclear Steam Supply System Scope: A preoperational piping vibrational and dynamics effects testing program was conducted for the reactor coolant loop/support system during startup functional testing of the plant. The purpose of these tests was to confirm that the system had been adequately designed and supported for vibration as required by Section III of the ASME Code, Paragraph NB-362.3. The preoperational piping vibration and dynamic effects test program for the primary coolant loop system (this includes the hot legs, cold legs, cross-over legs, RCPs, SGs, and reactor vessel) at South Texas Project Electric Generating Station (STPEGS) Units 1 and 2 was as follows:

1. The primary coolant loop system as defined above was instrumented with acceleromenters to measure the dynamic response of the system during normal and transient operating

conditions. In addition to normal steady state operation, the test conditions included steady-state operation with various combinations of RCPs in operation and transient conditions due to the starting and tripping of the RCPs.

2. The test data was analyzed to determine the maximum alternating stress induced in the piping due to the measure vibration. This alternating stress was compared to acceptance criteria based upon fatigue allowables for the piping. Piping was visually inspected to determine the acceptability of the steady-state vibrations. Vibration amplitudes were related to stress levels following the guidance of ANSI/ASME standard OM3, 1982, except as amended below. The piping was monitored by instrumentation at locations where vibrations appear to be excessive to demonstrate that the measured pipe deflections when coverted to stress would not exceed the following limits.

For steady-state vibration, the maximum calculated alternating stress intensity S_{alt} shall be limited as defined below:

(a) For ASME Class 1 piping systems:

$$S_{alt} = \frac{C_2K_2M}{Z} \leq \frac{S_{el}}{\alpha}$$

where:

C_2 = secondary stress index as defined in the ASME Code

α = allowable stress reduction factor: 1.3 for materials covered by Figure I-9.1; or 1.0 for materials covered by Figures I-9.2.1 or I-9.2.2 of the ASME Code

K_2 = local stress index s defined in the ASME Code

M = maximumzero to peak dynamic moment loading due to vibration only, or in combination with other loads as required by the system design specification

S_{el} = $0.8S_A$ where A_A is the alternating stress at 10^6 cycles from Figure I-9.1; or S_A at 1011 cycles from Figure I-9.2.2 of the ASME Code. Curves A, B, and C from Figure I-9.2.2 will be used per the criteria stated in that figure. The user shall consider the influence of temperature on the Modulus of Elasticity

Z = section modules of the pipe

(b) For ASME Class 2 and 3 piping, ANSI B31:

$$S_{alt} = \frac{C_2K_2M}{Z} \leq \frac{S_{el}}{\alpha}$$

where:

C_2K_2 = 2i

i = stress intensification factor, as defined in Subsection NC and ND of the ASME Code or B31

3. In the event that the measured vibration is found to be unacceptable based on the comparison with the acceptance criteria, appropriate corrective action will be implemented. This may consist of either:

- Further testing or analysis to demonstrate that the observed levels do not cause ASME stress and fatigue limits to be exceeded.

- Systems modification to eliminate the unacceptable vibration with subsequent test verification.

It should be noted that the layout, size, etc., of the RCL piping used in the STPEGS units are very similar to those employed in Westinghouse Electric Corporation (Westinghouse) plants now in operation. The operating experience that has been obtained from these plants indicates that the RCL piping is adequately designed and supported to minimize vibration. In addition, vibration levels of the RCP, which is the only mechanical component that could cause vibration of the RCL piping, are measured and held to the limits given in Section 5.4.1. Thus, excessive vibration of the RCL piping should not be present. However, as added assurance that excessive vibration is not present in the STPEGS units, the RCL system was subjected to the test program discussed above.

3.9.2.1.2 Balance-of-Plant Scope: Safety-related piping systems in BOP, are designed in accordance with the ASME Code, Section III. Each system is designed to maintain dynamic effects within acceptable limits. A preoperational test program as described in Section 14.2 was implemented as required by NB-3622.3, NC-3622, and ND-3611 of Section III of the ASME B&PV Code to verify that the piping and piping restraints would withstand dynamic effects due to transients such as pump trips and valve trips, and that piping vibrations are within acceptable levels.

The presoperational test program for the Class 1, 2, and 3 piping systems simulated actual operating modes to demonstrate that the appurtenances comprising these systems meet functional design requirements and that piping vibrations are within acceptable levels.

Piping systems were checked in three sequential series of tests and inspections. Construction acceptance, the first step, entailed inspections of components for correct installation. During this phase, pipe and equipment supports were checked for correct assembly and setting. The cold locations of RGS components, such as SGs and RCPs, were recorded.

During the second step of testing, plant heatup, the plant was heated to normal operating temperatures. During the heatup, systems were observed periodically to verify proper expansion and expansion data was recorded at the end of heatup.

During the third step of testing, performance testing, systems were operated and performance of critical pumps, valves, controls, and auxiliary equipment was checked. This phase of testing included transient tests, such s RCP trips, reactor trip, and relief valve testing. During this phase of testing, the piping and piping restraints were observed for vibration and expansion response. Automatic safety devices, control devices, and other major equipment were observed for indications of overstress, excess vibration, overheating, and noise. System tests included critical valve operation during transient system modes.

The locations in the piping system selected for observation during the testing, and the respective acceptance criteria, were provided in the detailed preoperational vibration, thermal expansion, and dynamic effects test program plan.

Provisions were made to verify the operability of essential snubbers by recording hot and cold positions. If vibration during testing exceed the acceptance criteria, corrective measures were taken and the test rerun to demonstrate adequacy.

Vibratory dynamic loading scan be placed in two categories; transient induced vibrations and steady-state vibrations. The first is a dynamic system response to a transient, time-dependent forcing function, such as fast valve closure, while the second is a constant vibration, usually flow induced.

1. Transient Vibrations

   Piping systems are designed in accordance with ASME Code to withstand dynamic transients due to rapid valve closing/opening, pump starts/stops, safety relief valve operation, etc. Dynamic transients which are found to be significant are analyzed by time-dependent dynamic analysis. The stresses thus obtained are combined with system stresses resulting from other operating conditions in accordance with the criteria provided in Section 3.9.1 and 3.9.3 to meet the ASME Code requirements.

   In order to provide additional assurance, piping systems discussed in Section 3.9.2.1 are tested for such dynamic transients as described above to corroborate the loads obtained by time-dependent dynamic analysis and to ensure that piping vibrations are within acceptable limits. The additional acceptance criteria for evaluating transient vibrations are as follows:

   a. Qualified test observers determine that the dynamic responses during the transient are not excessive, based on their past experience with similar systems.

   b. Qualified inspectors determine, following the transient, that there is no visible evidence of any structural deformation of the pipe or its restraint members.

   c. For systems which are instrumented for testing, instrumentation monitoring the dynamic response, if any, corroborates the judgement of the observers and inspectors by meeting predetermined acceptance criteria.

Systems failing to meet the above criteria are qualified by additional testing, dynamic analysis, change in operating procedures or additional restraints. Details of the program including the criteria for evaluation of data gained, are provided in the test procedures.

2. Steady-State Vibrations

   a. System vibration resulting from flow disturbances fall into this category. Positive displacement pumps may cause such flow variation and vibration. Pulsation dampers are provided on the suction and discharge sides of the positive displacement pumps to reduce the pressure pulsations.

b. Since the exact nature of the flow disturbance is not known prior to pump operation, no analysis is performed. If excessive system vibration is evidenced during initial operation, appropriate measures will be taken to reduce the vibration.

The acceptance criteria for steady state vibration testing is given in Section 3.9.2.1.1.

3.9.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment.

3.9.2.2.1 Safety-Related Nuclear Steam Supply System (NSSS) Mechanical Equipment: Westinghouse utilizes analysis, testing, or a combination of test and analysis to seismically qualify equipment. Testing is the preferred method; however, analysis is utilized when one of the following conditions is satisfied:

1. The equipment is too large or the external loads, connecting elements, or appurtenances cannot be simulated with a shaker table test.

2. The only requirement that must be satisfied relative to the safety of the plant is the maintenance of structural integrity (mechanical equipment only).

3. The component represents a simple linear system or nonlinearities can be conservatively accounted for in the analysis.

The operability of seismic Category I mechanical equipment must be demonstrated if the equipment is active; i.e., mechanical operation is relied on to perform a safety function. The operability of active safety class (SC) 2 and 3 pumps, active SC 1, 2, or 3 valves and their respective drives, operators and vital auxiliary equipment is shown by satisfying the criteria given in Section 3.9.3.2.

Inactive seismic Category I equipment such as heat exchangers (HXs), racks, and consoles are shown to have structural integrity during a seismic event by analysis satisfying the stress criteria applicable to the particular piece of equipment.

Chapter 13.7 describes alternate requirements for safety related non-risk significant and low safety significant structures, components, and systems.

A list of seismic Category I equipment and the method of qualification is provided in Tables 3.2.B-1 and 3.9-10.

3.9.2.2.2 Safety-Related Balance of Plant Mechanical Equipment: The operability of Category I mechanical equipment has been demonstrated for active equipment; i.e., mechanical operation is relied upon to perform a safety function. The operability of active Class 2 and 3 pumps, active Class 1, 2, and 3 valves, and their respective drives, operators, and vital auxiliary equipment has been shown by satisfying he criteria given in Section 3.9.3.2. Other active mechanical equipment is shown to be 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 Section 3.9.3.2 on pumps and valves. Testing procedures similar to the procedures outlined in Section 3.10 for electrical equipment have been used to demonstrate operability for the component which is mechanically or structurally so complex that its response cannot be adequately predicted analytically.

Inactive seismic Category I equipment has been shown to have structural integrity during all plant conditions in one of the following manners: (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.

Chapter 13.7 describes alternate requirements for safety related non-risk significant and low safety significant structures, components, and systems.

Safety-related seismic Category I BOP equipment and the method of qualification used are provided in Section 3.10.

3.9.2.3 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 indicate that the most important forcing functions are flow turbulence and pump-related excitation. The relevance of such excitations depends on many factors such as type and location of component and flow conditions. The effects of these forcing functions have been studied from test performed on models and prototype plants, as well as on component tests (Refs. 3.9-4, 3.9-5, and 3.9-6).

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 and the Sequoyah No. 1 plant have provided prototype data applicable to STPEGS Units 1 and 2 (Ref. 3.9-5, 3.9-6, 3.9-10, and 3.9-19).

The STPEGS Units 1 and 2 are similar to Indian Point No. 2; the only significant differences are the modifications resulting from the replacement of the annular thermal shield with neutron shielding pads, the change to the UHI-style inverted top hat support structure configurations, and the use of 17 x 17 extended length fuel. These differences are addressed below.

1. Neutron shielding pads lower internals

   The primary cause of core barrel excitation is flow turbulence generated in the downcomer annulus, which is not affected by the upper internals (Ref. 3.9-5). the vibration levels due to core barrel excitation for Trojan and STPEGS Units 1 and 2, both having neutron shielding pads, are expected to be similar. Since the downcomer annuli are similar (core barrel lengths differ by approximately 0.4 percent), the coolant inlet temperature of the STPEGS units is slightly different than Trojan 1 and the flow rate is slightly higher. Scale model tests show that core barrel vibration varies as velocity raised to a small power (Ref. 3.9-4). The difference in fluid density and flowrate result in a slightly higher core barrel vibration for STPEGS Units 1 and 2 than for Trojan 1. However, scale model test results (Ref. 3.9-4), and 2 than for Trojan 1. However, scale model test results (Ref. 3.9-4), and results from Trojan (Ref. 3.9-10) 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 stresses with large safety margins were measured at Indian Point No. 2 (thermal shield configuration) lead to the conclusion that stresses approximately equal to or less than those of Indian Point No. 2 are expected on the STPEGS Units 1 and 2 internals with the attendant large safety margins.

2. 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 and by pump related excitations (refs. 3.9-5 and 3.9-10). Sequoyah and STPEGS Units 1 and 2 have the same basic upper internals configuration; therefore, the general vibration behavior is not changed. The STEPEGS Units 1 and 2 upper internals adequacy has been determined from data from the instrumented plant test at Sequoyah 1, scale model tests and numerous operating plants. The results of testing at Sequoyah 1 (Ref. 3.9-19), showed that the components are excited by flow induced and pump related excitations. Analyses of the data indicate that the instrumented components have adequate factors of safety; random flow-induced responses are adequately predicted by scale models; and margins are higher with the core in place than during hot functional testing.

   In addition, the STPEGS Units 1 and 2 upper internals configuration was tested in a scale model using the same modeling techniques as for the scale model tests of the UHI configuration. The responses of the STPEGS Units 1 and 2 upper internals have been evaluated using the Sequoyah 1 and scale model information. The results show adequate factors of safety for all components.

3. 17 x 17 Fuel

   The only structural changes in the internals resulting from the design change from the 15 x 15 to the 17 x 17 extended length fuel assembly are the guide tube, control rod drive line, and lower core support structure. The new 17 x 17 guide tubes are stronger and more rigid, hence they are less susceptible to flow induced vibration. The 17 x 17 extended length fuel assembly has the same basic frequency characteristics as the 15 x 15 fuel assembly. The added length is accommodated as shown in Figure 3.9-1 resulting in essentially the same core barrel length. Thus, the STPEGS Units 1 and 2 lower internals vibration behavior is expected to be very similar to the vibration with 15 x 15 fuel assemblies.

   The original test and analysis of the four-loop configuration is augmented by References 3.9-4, 3.9-5, 3.9-6, 3.9-10 and 3.9-13 to cover the effects of successive hardware modification. Also, analytical studies have been performed which demonstrate that the lower core support structure modifications to accommodate the 17 x 17 extended length fuel have no significant effect on the core barrel vibration.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals. Because the STPEGS Units 1 and 2 reactor internals design configuration is well characterized, as was discussed in Section 3.9.2.3 it is not considered necessary to conduct instrumented tests of the STPEGS hardware. The requirements of RG 1.20 will be met by conducting the confirmatory preoperational testing examination for integrity in accordance with RG 1.20. This examination will included some 35 points with special emphasis on the following areas.

1. All major load-bearing elements of the reactor internals relied upon to retain the core structure in place.

2. The lateral, vertical, and torsional restraints provided within the vessel.

3. Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals.

4. Those other locations on the reactor internal components which are similar to those which were examined on the prototype designs.

5. The inside of the vessel will be inspected before and after the hot functional test, with all internals removed, to verify that no loose parts or foreign material are present.

A particularly close inspection will be made on the following items or areas using a 5X or 10X magnifying glass or penetrant testing, where applicable.

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 any 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 shield panel screw locking devices and dowel pin locking device. Examine the interface surfaces for evidence of tightness and for locking device integrity.

8. Radial support key welds.

9. Insert screw locking devices. Examine soundness of lockwelds

10. Instrumentation guide tubes. Check all 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 for any 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. One would expect to see, on the bearing surfaces of the key and keyway, burnishing, buffing, or shadow marks which would indicate pressure loading and relative motion between the two. One would expect to see, on the bearing surfaces of the key and keyway, burnishing, buffing, or shadow marks which would indicate pressure loading and relative motion between the two parts. Some scoring of engaging surfaces is also possible and acceptable.

13. Gaps at baffle joints. Check for unacceptable gaps between baffle and top former and at baffle to baffle joints.

<u>Upper Internals</u>

1. Thermocouple conduits, clamps, and couplings.

2. Guide tube, support column, orifice plate, and thermocouple assembly locking devices.

3. Upper core plate alignment inserts. Examine for any shadow marks, burnishing, buffing, or scoring. Check for locking devices for integrity of lockwelds.

4. Thermocouple conduit clamp welds.

5. Guide tube enclosure 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) for 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 and 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.

3.9.2.5 <u>Dynamic System Analysis of the Reactor Internals Under Faulted Conditions</u>. The following events are considered in the faulted conditions category:

1. LOCA (RCL branch pipe ruptures are considered.)

2. SSE

Maximum stress for SSE and LOCA are obtained and combined.

Maximum stress intensities are compared to allowable stresses for the faulted condition. Elastic analysis is used to obtain the response of the structure, and the stress analysis of each component is performed according to ASME Code-approved techniques. For faulted conditions, stresses are above yield in a few locations. For these cases only, some inelastic stress limits are applied.

The design rules of Subsection NG of the ASME B&PV Code, Section III, apply to those reactor internals components identified as 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 for the internals are concerned with the deflections and stability of the parts in addition to stress criteria to assure integrity of the components.

For the critical internal structures, maximum allowable defections, based on functional performance criteria, are listed in Table 3.9-9. The basic operational or functional criterion to be met for the reactor internals is that the plant shall be shut down and cooled in an orderly fashion so that fuel-cladding temperature is kept within specified limits following a Design Basis Accident (DBA).

Reactor Internals Analysis

The evaluation of the reactor internals is composed of two parts. The first part is the three-dimensional response of the reactor internals resulting from the RCL branch pipe break conditions mentioned in Section 3.9.1.4.6.1. The reactor internals response is taken from the WECAN RPV and internals system response as described in Section 3.9.1.4.6.4 for the RPV support analysis. The second part of this evaluation is the core-barrel shell response which consists of the various N = 0, 2, 3, etc., ring mode response occurring in the horizontal plane. This second part, or ring mode evaluation, is independent of the loop forces.

Analysis of the reactor internals for blowdown loads resulting for an RCL branch pipe break is based on the time-history applied blowdown forcing functions. The forcing functions are defined at points in the system where changes in cross section of direction of flow occur in such a way that differential loads are generated during the blowdown transient. The dynamic mechanical analysis can employ the displacement method, lumped parameters, and 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.

MULTFLEX a blowdown digital computer program (Ref. 3.9-7), which was developed for the purpose of calculating local fluid pressure, flow, and density transients that occur in pressurized water reactor coolant systems during a LOCA, is applied to the subcooled region and which, due to their method of solution, could not be extended into the region in which large changes in the sonic velocities and fluid densities take place. MULTIFLEX 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, and expansion, as well as some effects of the water/solid interaction, are considered.

The MULTFLEX code evaluates the pressure and velocity transients for a maximum of 2,400 locations throughout the system. Each reactor component for which calculations are required is designated as an element and assigned an element number. Forces acting upon each of the elements are calculated, summing the effects of:

1. The pressure differential across the element

2. Flow stagnation on, and unrecovered orifice losses across, the element

3. Friction losses along the element

Input to the calculation 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 the shear forces act.

The reactor internals analysis has been performed using the following assumptions:

- The analysis considers the effect of hydroelasticity.

- The reactor internals are represented by concentric pipes, beams, concentrated masses, linear and nonlinear springs, and dashpots simulating the nonlinear response of the components

- 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.

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 1-millisecond time is taken as the limiting case.

In the case of a hot leg branch pipe break, 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, are possible responses of the barrel during hot leg blowdown. In addition to the above effects, the hot leg break results in transverse loading on the upper core components as the fluid exits the hot leg nozzle.

In the case of a cold leg branch pipe break, a rarefaction wave propagates along a reactor inlet pipe, arriving first at the core barrel at the inlet nozzle of the affected 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 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 LOCA, the combined effect of the maximum stresses for each case is considered. In general, the loading imposed by the earthquake is small compared to the blowdown loading. The seismic analysis of the reactor internals is discussed in Section 3.7.3.

A summary of the analysis for major components is presented in the following paragraphs. Reference 3.9-9 provides the basic methodology used in the reactor internals blowdown analysis.

1. Core Barrel

   For the hydraulic analysis of the pressure transients during hot leg branch pipe 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:

   a. The effect of the fluid environment is neglected.

   b. The shell is treated as simply supported.

   During a cold leg branch pipe blowdown, the upper barrel is subjected to a nonaxisymmetric 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 a cold leg branch pipe blowdown is performed as follows:

   a. The core barrel is analyzed as a shell with two variable sections to model the core barrel flange and core barrel.

   b. The barrel with the core and neutron shielding pads is analyzed as a beam elastically supported at the top and at the lower radial support, and the dynamic response is obtained.

2. Guide Tubes

   The dynamic loads on rod cluster control (RCC) guide tubes are more severe for a LOCA caused by hot leg branch pipe rupture than for an accident caused by cold leg branch pipe rupture, since the cold leg break leads to much smaller changes in the transverse coolant flow over the rod cluster control assembly (RCCA) guides. The guide tubes in closet proximity to the outlet nozzle for a hot leg branch pipe break are the most severely loaded. The transverse guide tube forces during a blowdown decrease with increasing distance from the ruptured nozzle location.

A detailed structural analysis of the RCC guide tubes is performed to establish the equivalent cross section properties and elastic end support conditions. An analytical model is verified by subjecting the RCC guide tube to a concentrated force applied at the midpoint of the lower guide tube. In addition, the analytical model has been previously verified through numerous dynamic and static tests performed on the 17 x 17 guide tube design.

The response of the guide tubes to the transient loading from blowdown resulting from hot leg branch pipe breaks is found by representing the guide tube as an equivalent three-dimensional beam in which each node of the beam has six degrees of freedom.

3. Upper Support Columns

Upper support columns located close to the nozzle of the affected hot leg will be subjected to transverse loads due to cross flow. The loads applied to the columns are computed wit 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 sections and the resulting stresses are obtained using the reduced section modulus and appropriate stress risers for the various sections.

4. Results of Reactor Internals Analysis

Maximum stresses due to the SSE (vertical and horizontal components) and a LOCA were obtained and combined. All core support structure components were found to be within acceptable stress and deflection limits for both hot leg and cold leg branch pipe LOCAs occurring simultaneously with the SSE; the stresses and deflections which would result following a faulted condition are less than those which would adversely affect the integrity of the core support structures. For the transverse excitation, it is shown that the barrel does not buckle during a hot leg branch pipe break and that it meets the allowable stress limits during all specified transients.

The results obtained from linear analyses indicate that the relative displacement between the components will close the gaps, and consequently the structures will impact on each other. Linear analysis will not provide information about the impact forces generated when components impact on each other; however, in some instances, linear approximations can and are applied prior to and after gap closure. The effects of the gaps that could exist between vessel and barrel, between fuel assemblies, and between fuel assemblies and baffle plates, were considered in the analysis using both linear approximations and non-linera techniques. Both static and dynamic stress intensities are within acceptable limits.

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 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.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results. As stated in Section 3.9.2.4, it is not considered necessary to conduct instrumented test of the STPEGS RPV internals, as their adequacy has been verified by use of the Sequoyah and Trojan results as well as by the 1/7 scale model test results. References 3.9-5 and 3.9-10 describe predicted vibration behavior based on studies performed prior to the plant tests. These studies, which utilize 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 characterize the forcing functions and establish component structural characteristics so that the flow-induced vibratory behavior and response levels for STPEGS are estimated. These estimates are then compared to values deduced from plant test data obtained from the Sequoyah and Trojan internals vibration measurement programs.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits. The load combinations and the design stress limits associated with the plant operating conditions which are applied to the design and analysis of components and component supports are defined herein. The plant conditions considered were normal operation, postulated accidents, and specified seismic events. Design transients are further discussed in Section 3.9.1.

3.9.3.1.1 Loading Combinations, Design Transients, and Stress Limits for NSSS Components and Supports: The ASME Code Class components are constructed in accordance with the ASME B&PV Code, Section III requirements. For Code Class 1 components, very stringent requirements are imposed and are met. For Code Class 2 and 3 components, the requirements are less stringent but sufficiently conservative, in accordance with the lower classification.

3.9.3.1.1.1 ASME Code Class 1 Components – Loading considered for ASME Class 1 components and supports are presented in Table 3.9-2.1 and stress limits for these components are given in Table 3.9-2.1A. A detailed discussion of design transient for NSSS components is provided in Section 3.9.1.

3.9.3.1.1.2 ASME Code Class 2 and 3 Components – The design loadings considered for ASME Code Class 2 and 3 components and supports furnished with NSSS are given in Table 3.9-2.2.

The allowable 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 Tables 3.9-3A, 3.9-4B, 3.9-4C, and3.9-6. Active[(1)] pumps and valves are further discussed in Section 3.9.3.2. The component supports are designed in accordance with ASME B&PV Code, Section III, Subsection NF (Section 3.9.3.4).

3.9.3.1.2 Loading Combinations, Design Transients, and Stress Limits for Balance-of-Plant Components and Component Supports: Design pressure, temperature, and other loading conditions that provide the bases for design of fluid systems Code Class 2 and 3 components are presented in the

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[1] Active components are those whose operability is relied upon to perform a safety function (as well as reactor shutdown function) during the transients or events considered in the respective operating condition categories.

sections which describe the systems. Refer to Section 33.9.1.4.7 for discussion of stress criteria of ASME Code Class 1 components. Refer to Table 3.9-2.5 for conformance with RG 1.48

3.9.3.1.2.1 Design Loading Combinations – The design loading combinations for ASME Code Class 1 BOP components and component supports are given in Tables 3.9-2.3 and 3.9-2.4. The design loading combinations for ASME Code Class 2 and 3 BOP components and component support are given in Tables 3.9-2.3A and 3.9-2.4.

3.9.3.1.2.2 Design Stress Limits – The design stress limits established for components are sufficiently low to ensure that violation of the pressure retaining boundary does not occur and that the components operate as required. Stress limits for Class 2 and 3 components for each of the loading combinations are component-oriented and are presented in Tables 3.9-3, 3.9-4, 3.9-4A, 3.9-6A, 3.9-7A, and 3.9-7C for vessels, nonactive pumps, active pumps, valves, piping, and component supports, respectively. Active pump and valve operability are discussed in Section 3.9.3.2. The component supports are designed in accordance with ASME Code, Section III, Subsection NF, and are discussed in Section 3.9.3.4.

3.9.3.1.2.3 Applicable Codes And Standards – The following codes and standards are used as a basis for the piping design which includes piping stress analysis and the design, fabrication, construction and testing of the pipe supports. Different issue dates of these documents may be used provided they meet the minimum requirements stated herein. Code cases and other standards are given in project design criteria and design specifications and in response to Q210.07N.

1. American Society of Mechanical Engineers (ASME) – ASME B&PV Code, Section III Subsections NA, NC, ND and NF, 1974 Edition, including Winter 1975 is used for piping stress analysis and piping support design. Other addenda used for piping stress analysis of Class 2 and 3 systems are: Paragraph NC/ND3611.2 of Winter 1976 Addenda, NC3652.3 of 1977 Edition of code, NC/ND3622.5 of Winter 1978 Addenda and NC/ND3658.3 of Summer 1979 Addenda, NC-3652 through NC-3655 of Winter 1981 Addenda, ND-3652 through ND-3655 of Summer 1984 Addenda. Containment mechanical penetrations have been designed in accordance with the requirements of NC-3200 and NC-3600 of the ASME Code, Section III, 1974 Edition with Addenda through Summer 1976 and NC-3217 of the Winter 1976 Addenda.

2. American National Standards Institute (ANDI) – ANSI B31.1 Power Piping code, 1973 Edition including Winter 1975 Addenda.

3. American Institute of Steel Construction (AISC)

   A. “AISC – Specification for the Design, Fabrication and Erection Structural Steel for Buildings”, 1969 Edition, including supplements 1, 2, and 3.

   B. “Specification for Structural Joints using ASTM A325 or ASTM A490 Bolts”, 1969, including 1976 Addendum.

   C. “Code of Standard Practice for Steel buildings and Bridges”, 1972.

4. Manufacturers Standardization Society (MSS) MSS-SP-58-1975, “Hangers and Supports-Materials, Design and Manufacture”.

5. MSS-SP-69-1976, “Pipe Hangers and Supports – Selection and Application”.

6. American Welding Society (AWS) AWS-A2.4-1979, “Symbols for Welding and Non-Destructive Testing”.

7. ANSI A58.1-1972, “American National Standard Building Code Requirements for Minimum Design Loads in Buildings and Other Structures”.

3.9.3.2 Pump and Valve Operability Assurance. Mechanical equipment classified as safety-related has been shown to be capable of performing its function during the life of the plant under postulated plant conditions. Equipment with faulted condition function requirements include active pumps and valves in safety-related fluid systems. Seismic analysis is represented in Section 3.7 and covers all safety-related mechanical equipment. A list of active components is presented in Tables 3.9-1.1, 3.9-1.2, and 3.9-1.2A. Inservice inspection (ISI) and testing are discussed in Sections 5.2.4 and 6.6.

Chapter 13.7 describes alternate requirements for safety related non-risk significant and low safety significant structures, components, and systems. These alternate requirements may be applied to both BOP and NSSS equipment.

To assure operability, the following items are included in the design specifications of active BOP pumps and valves:

1. For BOP pumps nozzle loads (external piping loads) are specified in the pump design specifications. These loads are considered by the vendor in the deformation analysis or testing of the pump. It is stated in the design specification that the vendor shall demonstrate the operability of the equipment under these loading conditions. Loads on the pump die to the connected piping are kept within these specified nozzle loads.

2. For active BOP valves, valve end loads (external piping loads) are specified in the valve design specification. These loads are considered by the vendor in the deformation analysis or testing of the valve. It is stated in the design specification that the vendor shall demonstrate the operability of the active valves when subjected by these valve end loads. Loads on the valve due to the connected piping are kept within these specified end loads.

3. It is specified in the equipment specification that all pumps and valves should perform their intended safety function before, during, and after the seismic disturbances and during the specified post-accident environment including radiation dose, pressure, humidity, and temperature.

4. Deformation limits and operating clearances have been specified by the equipment manufacturer and it is stated in the specification that excessive rubbing on rotating parts is not acceptable for active pumps under the accident conditions.

5. It is stated in the specification that the vendor should demonstrate the ability of the equipment no to suffer any loss of function during and after the specified conditions.

3.9.3.2.1 Pump and Valve Operability Assurance (NSSS Scope): Mechanical equipment classified as safety-related must be capable of performing its function under postualted plant conditions. Equipment with faulted condition functional requirements includes active pumps and valves in fluid systems such as the Residual Heat Removal System (RHRS), Safety Injection System (SIS), and Containment Spray System (CSS). While seismic analysis demonstrates the structural integrity of the active pump and valve assembly, operability is assured by satisfying the requirements of the Westinghouse Pump and Valve Operability Program. Through this program, operability of the active mechanical equipment is demonstrated by test or a combination of test and analysis where the analysis has been supported by testing. The tests have been performed on prototype equipment and similarity analysis is then used to justify applicability to plant specific equipment. Design integrity of the pump and valve is demonstrated by complying with the GDC-4 requirements for the mechanical portion of the equipment and the electrical portion of the assembly is qualified in accordance with 10CFR50.49 (e.g., active valve appurtenances such as limit switches, etc. are qualified by separate testing). Operability of the entire assembly is demonstrated. Active pumps and valves within Westinghouse scope are identified in Tables 3.9-1.1 and 3.9-1.2A, respectively. Seismic analysis covering safety-related mechanical equipment is presented in section 3.7. ISI and testing are discussed in Section 5.2.4 and 6.6.

3.9.3.2.1.1 Pump Operability Program (NSSS Scope) – All active pumps are qualified for operability by first undergoing rigid tests prior to 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 times the ratio of material allowable stress at room temperature to the allowable stress value at the design temperature; (2) seal leakage tests; and (3) performance tests to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump parameters. Also monitored during these operating tests are bearing temperatures (Section XI tested pumps only) and vibration levels. Bearing temperature limits are determined by the manufacturer based on the bearing material, clearances, oil type and rotational speed. These limits are approved by Westinghouse. After the pump is installed in the plant preservice and inservice testing further supplement the qualification process. After the pump is installed in the plant it undergoes preoperational and power ascension testing, and inservice testing and inspection to ensure operability during plant 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 by assuring that they will start up, continue operating, and not be damaged during the faulted conditions. the pump manufacturer are required to show by analysis correlated by tests, prototype tests, or existing documented data that the pump will perform its safety function when subjected to loads imposed by the maximum seismic accelerations and the maximum faulted nozzle loads. It is required that testing or dynamic analysis be used to show that the lowest natural frequency of the pump is greater than 33 Hz. This frequency is sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A pump with a natural frequency above 33Hz is considered essentially rigid. A static shaft deflection analysis of the rotor is performed with the conservative SSE accelerations of 2.1g in two orthogonal horizontal directions and of 2.1g in the vertical, acting simultaneously. The defections 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. If rubbing or impact is predicted, prototype tests or existing documented data is used to demonstrate that the pump will not be damaged or cease to perform its design function; the effect of rubbing or impacting on pump operation is evaluated by comparison of the contacting surfaces to similar surfaces of pumps that have been tested.

In order to avoid damage during the faulted plant condition, the stress levels caused by the combination of normal operating loads, SSE, and dynamic system loads are restricted to the limits indicated in Table 3.9-4B. In addition, the pump casing stresses caused by the maximum faulted nozzle loads are restricted to the stresses outlined in Table 3.9-4B. The changes in operating rotor clearances caused by casing distortions due to these nozzle loads are considered. The maximum seismic nozzle loads combined with the loads imposed by the seismic accelerations are also considered in an analysis of the pump supports. Furthermore, the calculated misalignment is shown to be less than that misalignment which could cause pump misoperation. The stresses in the supports are below those in Table 3.9-4B; this ensures that support distortion is of short duration (equal to the duration of the seismic event) and support elasticity is maintained.

Performing these analyses with the conservative loads stated, and with the restrictive stress limits of Table 3.9-4B as allowables, assures that critical parts of the pump will not be damaged during the short duration of the faulted condition and that, therefore, the reliability of the pump for post-faulted condition operation will not be impaired by the seismic event.

If the natural frequency is found to be below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis and adjusted accelerations are determined using the same conservatisms contained in the accelerations used for “rigid” structures (2.1g orthogonal horizontal and vertical). The static analysis is performed using the adjusted accelerations; the stress limits stated in Table 3.9-4B must still be satisfied.

To verify analytical techniques and provide data for correlation to analytical results, full assembly operability testing was performed on a Charging/Safety Injection Pump. The assembly consisted of an 11-stage centrifugal pump, a speed increaser gear and a 600 hp induction motor mounted on a common base-plate typical of normal plant installation. Of all Westinghouse supplied NSSS active pump assemblies, this one was chosen as being most representative of the various design features of active pumps. The assembly was mounted on a shaker table such that triaxial seismic input could be simulated. A flow loop connected to the pump permitted full pump operation while special fixtures were fabricated to apply nozzle loads to the suction and discharge nozzles. Instrumentation including accelerometers, strain gauges, strain bolts, proximity probes and thermocouples were used to monitor the complete assembly during testing.
In general, the testing consisted of a preseismic resonance search, a preseismic pump head-flow characterization, five OBEs, four SSEs, pump head-flow characterization between seismic runs, a post-test resonance search and a post-test pump head-flow characterization. The pump was started prior to and during seismic testing without difficulty. As a result of the testing, no pump damage was visually observed or measured and the hydraulic characteristics remained within specific tolerances. It was concluded that the test pump assembly remained operational during and after a design basis seismic event.

The specific pump attributes (e.g., weight, RPM, gear ratio, full load current) of both the test unit and the pumps employed at the STPEGS are compared in a Pump and Valve Operability report which includes a summary report for the testing performed on the charging pump assembly.

To complete the seismic qualification procedures, the pump motor is qualified for operation during the maximum seismic event. Any auxiliary equipment identified as vital to the operation of the pump or the pump motor and which is not proven adequate for operation by the pump or motor qualifications are separately qualified by meeting the requirements of IEEE 344-1975, with the additional requirements and justifications outlined in Section 3.9.3.2.1.3.

The program described 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.

3.9.3.2.1.2 Valve Operability Program (NSSS Scope) – Safety-related active valves must perform their safety-related mechanical motion in times of an accident. Tests and analyses are conducted to provide assurance that these valves will operate during a seismic event.

The safety-related valves are subjected to a series of stringent tests prior to service and during 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 valves will open and close within the specified time limits when subjected to the design differential pressure. After the valve is installed in the plant preservice and inservice testing further supplement the qualification process. After the valve is installed in the plant it undergoes preoperational and power ascension testing, and inservice testing and preoperational and power ascension testing, and inservice testing and inspection to ensure operability during plant operation. Periodic inservice inspections, and periodic inservice operation 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. Compliance with RG 1.148 is described in Table 3.9-23 and 3.9-24.

Active valves are designed in accordance with ASME B&PV Code Section III. To demonstrate structural integrity, an analysis of the valve extended structure is performed for static equivalent seismic SSE loads applied at the center of gravity of the extended structure. The maximum stress limits used for active Class 2 and 3 valves are shown in Table 3.9-6. Class 1 valves are designed/analyzed according to the rules of the ASME Code, Section III, NB-3500.

In addition to these tests and analyses, full assembly valves, representative of each design type undergo tests to verify operability during a simulated plant faulted condition event by demonstrating operational capabilities within the specified limits. Westinghouse, working in conjunction with the valve manufacture, evaluates the various valve attributes (e.g., material composition, weight, wall thickness, size) and selects valves that are the most susceptible to seismic induced loads for testing. This permits extrapolation of demonstrated operational performance to other valves within the design family. The pump and valve operability report prepared for this project identifies the tested valves (including valve sizes) and each plant specific valve for which a given tested valve is employed for qualification purposes. A comparative analysis is performed with conclusions drawn on acceptability. The test procedures are described below.

The valve is mounted in a manner that conservatively represents typical valve installations. The valve includes the operator and appurtenances normally attached to the valve in service. The faulted condition nozzle loads are considered in either of two ways: (1) loads equivalent to the faulted condition nozzle loads are simultaneously applied to the valve (through its mounting) during the test; or (2) by analysis, the nozzle loads are shown to not affect the operability of the valve. Operability of the valve during a faulted condition is demonstrated by satisfying the following criteria:

1. Active valves are designed to have the lowest natural frequency greater than 33 Hz.

2. The complete valve assembly extended structure is statically deflected by an amount equal to the deflection caused by the faulted condition accelerations by applying the appropriate loads representing these accelerations at the center of gravity of the extended structure in the direction that yields the greatest deflection. The design pressure of the valve is simultaneously applied to the valve during the static deflection tests.

3. The valve is cycled while in the defected position, and cycle times are recorded. This data is compared to similar data taken in the undeflected condition to evaluate the significance of any change.

4. Motor operators, and other appurtenances necessary for operation are qualified by IEEE 344-1975 with additional requirements and justifications as supplied in Section 3.9.3.2.1.3.

The accelerations which are used for the static valve qualification shall be equivalent, as justified by analysis, to 4.0g acting in two orthogonal horizontal directions and 4.0g vertical, simultaneously. The piping designer must limit accelerations to these levels.

If the lowest natural frequency of the valve is less than 33 Hz, a dynamic analysis is performed to determine the equivalent acceleration to be applied during the static test. The analysis accounts for the amplification of the input acceleration, by considering the natural frequency of the valve and the frequency content of the applicable plant floor response spectra. The adjusted accelerations are determined using the same conservatisms contained in the 4.0g horizontal and 4.0g orthogonal vertical accelerations used for “rigid” valves. The adjusted acceleration is then used in the static analysis, and valve operability is assured by the methods outlined.

Valves that are safety-related but can be classified as not having an extended structure, such as check valves and safety vales, are considered separately. Check vales 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 that 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 design or analyses methods, the ability of the valve to operate is assured by the design features. The valve also undergoes the following: (1) stress analysis of critical parts which may affect operability, including the faulted condition loads (2) in-shop hydrostatic test,

(3) in-shop seat leakage test, and (4) periodic in situ valve exercising and inspection to assure functional ability of the valve.

Pressurizer safety valves are qualified by the following procedures (these valves are also subjected to tests and analysis similar to check valves): (1) stress and deformation analyses of critical items that might affect operability for faulted condition loads, (2) in-shop hydrostatic and seat leakage tests, and (3) periodic in situ valve inspection. In addition, 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 over pressurization safety capabilities during a seismic event.
Using these methods, all safety-related valves in the systems are qualified for operability during a faulted event. The methods outlined above conservatively simulate the seismic event and assure that the active valves will perform their safety-related function. Alternate valve operability testing, such as dynamic vibration testing is allowed if it is shown to adequately assure the faulted condition functional ability of the valve system.

3.9.3.2.1.3 Pump Motor and Valve Operator Qualification (NSSS Scope) – Motors for active pumps and motor operators for active valves and all vital electrical appurtenances thereto, are seismically qualified in accordance with IEEE 344-1975. If the testing option is chosen, sine-beat testing is justified. This justification may be provided by satisfying one or more of the following requirements to demonstrate that multi-frequency response is negligible or that the sine-beat input is of sufficient magnitude to conservatively account for this effect.

1. The equipment response is basically due to one mode.

2. The sine-beat response spectra envelops the floor response spectra in the region of significant response.

3. The floor response spectra consists of one dominate mode and has a peak at this frequency.

If the degree of coupling in the equipment is small, then single-axis testing is justified. Multi-axis testing is required if there is considerable cross-coupling; however, if the degree of coupling can be determined, then single-axis testing ca be used with the input sufficiently increased to include the effect of coupling on the response of the equipment.

Seismic qualification by analysis alone, or by a combination of analysis and testing, may be used when justified. The analysis program can be justified by demonstrating: (1) that equipment being qualified is amenable to analysis, and (2) that the analysis be correlated with tests or be performed using standard analysis techniques.

3.9.3.2.2 Pump Operability (BOP Scope): Safety-related active pumps are qualified by in-shop tests as appropriate for each type of pump and seismic qualification prior to installation in the plant. The in-shop tests include: (1) hydrostatic tests of pressure-retaining parts to 150 percent of the design pressure and (2) performance tests which are conducted while the pump is operated with flow to determine total developed head, minimum and maximum head, NPSH requirements, and other pump/motor properties. Where appropriate, bearing temperatures and vibration levels are monitored during these operating tests. After the pump is installed in the plant, startup tests are conducted. A range of operating temperatures is experienced during the power ascension stage. The required periodic inservice inspection and operational testing are performed. These tests demonstrate that the pump ill function as required during all normal operating conditions for the design life of the plant. The post-accident operating conditions for safety-related pumps do not differ significantly form normal operating condition. The range of temperature, NPSH, and flow experienced by each pump during preoperational testing, normal operation and inservice testing is similar to post-accident conditions. In addition to the above tests, the operability during the seismic event is shown by one of the following programs:

1. An individual pump, selected as a prototype, has been tested in the manufacturer's shop, with the test conditions equivalent to the combined plant conditions which the pump is expected to withstand at the time the active function is required. Vibratory excitation of the pump to simulate seismic loading is demonstrated: (a) by a separate test under conditions sufficiently severe to provide adequate margins for assurance of operability under combined plant loading conditions; or (b) by seismic analysis of critical pump components.

2. An individual pump, selected as a prototype, has been tested partially: (a) in the manufacturer's shop under those test conditions as limited by the test facility, (e.g., hydrostatic tests, seat leakage test, and performance test [also during these tests, bearing temperature and vibration levels have been monitored]); (b) in a testing laboratory for simulated seismic excitation loadings; and (c) in the plant after pump installation for confirmation of operability under flow conditions during system preoperational hot functional tests.

3. Pumps which are equivalent to a prototype pump that has successfully met the test requirements of a pump operability assurance program, are not tested if the loading conditions for those pumps are equivalent to or less than those imposed during testing of the prototype pump.

   The test results of the prototype pump are documented according to ANSI N45.2, Section 18.

   The prototype pump is selected from a group of similar pumps which are used in the plant. A prototype pump used in one nuclear power plant is deemed to qualify as a prototype pump for other plants provided that the system operating conditions of both plants and the pump loading conditions at the time when the active function is required are equivalent or less severe.

The pump manufacturer is required to show by testing, analysis, or existing document data that the pump will perform its safety function when subjected to the maximum seismic accelerations and maximum faulted nozzle loads. The pumps are tested or analyzed for the lowest natural frequency. The pump, when having a natural frequency above 33 Hz, is considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft defection analysis of the rotor is performed using the zero period acceleration (ZPA) of the applicable seismic response spectra in two orthogonal horizontal directions and in the vertical direction simultaneously. The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. If rubbing or impact occurs, its duration must be short and shown by prototype test or existing documented data not to unacceptably damage or prevent the pump from performing its design function. In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, and dynamic system loads are kept limited to the material elastic limit, as indicated in Table 3.9-4A. The maximum seismic nozzle loads are considered in an analysis of the pump supports to assure that a system misalignment cannot occur.

In cases where the natural frequency is found to be below 33 Hz, a dynamic analysis has been performed using the applicable seismic response spectra. The deflections determined from the analysis are compared to the allowable rotor clearances.

Faulted nozzle loads are provided in the pump design specification. External piping loads on the pump nozzles are kept within these specified limits. The pump specification requires the vendor to demonstrate the operability of the pump when subjected to the load combinations given in Table 3.9-2.3A. In addition, the pump casing stresses resulting from the maximum faulted nozzle loads are limited to the values given in Table 3.9-4A.

Environmental service conditions for normal, abnormal and accident conditions are identified in Section 3.11. Safety-related active pumps are environmentally qualified for operability during conditions where their operation is essential.

Performing these analyses with the conservative loads stated and with the restrictive stress limits as allowables assures that critical parts of the pump do not get damaged during the faulted conditions; therefore; therefore, the reliability of the pump for post-faulted condition operation is not impaired by the seismic event.

To complete the qualification procedures, the pump motor has been qualified for operation during the maximum seismic event. Any auxiliary equipment which is vital to the operation of the pump motor qualification has been separately qualified by meeting Institute of Electrical and Electronics Engineers (IEEE) 344-1975.

Similarity is established between the prototype and a group of pumps by pumps by virtue of the following characteristics.

1. Manufacturer – Pumps should be from the same manufacturer.

2. Geometry and Structure – Pumps should be of same type, size and physical characteristics.

3. Hydraulic Rating – Pumps should be of same capacity and head.

Operability of the pump is verified by analysis by assuming that the rotor of the pump does not interfere with the casing while rotating. Deflection of rotor is maintained within certain tolerance such that operation of the pump and its hydraulic characteristics remain unchanged. Deflection of the rotor depends upon the stiffness of shaft, bearing, pedestal and the body of the pump.

The qualification of pump and pump driver as an assembly is performed by analysis and/or by testing. In cases where the pump and the driver are qualified separately by either analysis or testing, the coupling between the components is analyzed to demonstrate the misalignment does not occur.

3.9.3.2.3 Valve Operability (BOP Scope): Safety-related active valves are subjected to a series of stringent tests prior to service and during plant life. The following tests are performed on active valves (except check valves): (1) shell hydrostatic test or ASME Section III requirements, (2) seat leakage test or disc hydrostatic tests, and (3) operational tests to verify that active valves except check valves will open and close within the specified time limits. For qualification of motor operators for environmental conditions, refer to Section 3.11. Cold hydrostatic-tests, periodic inservice inspections, and periodic inservice operations are performed in situ to verify the functional ability of the valve. A range of operating temperatures is experienced during the power ascension stage. With required periodic maintenance, these tests demonstrate reliability of the valves for the design life of the plant. Compliance with RG 1.148 is addressed in Tables 3.8-23 and 3.9-24.

The operability of active valves, including valve operators, under plant conditions when their respective safety function is relied upon to effect either a plant shutdown or to mitigate the consequences of an accident, has been demonstrated to the extent of availability and capability of test equipment by any one of the following acceptable programs:

1. An individual valve, selected as a prototype valve, has been tested with the test conditions imposed during the demonstration of valve opening and/or closing equivalent to the combined plant conditions (pressure, SSE, nozzle loads) that the valve is expected to withstand at the time the active function is required. (Such a test program is done for valves with a maximum size of 6 inches.)

2. An individual valve, selected as a prototype valve, has been tested under conditions which simulate separately each of the plant loadings (including SSE seismic loadings) that the valve is expected to withstand in combination during valve opening and/or closing.

   Sometimes such a test program has been supplemented by analyses which demonstrate that the individual test loadings are sufficiently higher than the plant loadings, to provide adequate margins for assurance of operability under combined loading conditions.

3. An individual valve, selected as a prototype, has been tested partially: (a) in the manufacturer’s shop under those test conditions as limited by the test facility (e.g., shell hydrostatic test, back-seat and main seat leakage tests, disc hydrostatic test), and operational tests to verify the opening and closing of the valve; (b) in a testing laboratory for simulated seismic excitation loadings; and (c) in the plant after valve installation for confirmation of operability under flow conditions during system preoperational hot functional tests.

   The test results of the prototype valve are documented according to ASNI N45.2, Section 8.

   The prototype valve is selected from a group of similar valves which are used in the plant. A prototype valve used in one nuclear power plant is deemed to qualify as a prototype vale for another plant provided the system operating conditions of both plants and the valve loading conditions at the time when the active function is required are equivalent or less severe.

4. When valves have been qualified by similarity analysis they are similar to a valve which has been already qualified by test or combination of test and analysis. Following are the characteristics considered in determining that a valve is similar to the tested prototype valve and forms the technical basis for qualification by similarity:

   a. Manufacturer – Valves are from the same manufacturer.

b. Geometry and Structure – Valves are of the same type and configuration. A valve is not considered similar to a qualified valve if the ratio of the sizes between the valve and qualified valve is greater than 1.5.

c. Pressure Rating – In general, a valve of lower pressure rating is selected for qualification and extended to the valves of higher pressure rating.

The mathematical model used in the prototype valve has been checked to agree with the experimental results. A substantially similar mathematical model has been used for the other valve with a slight change in the size of the operator and/or size of the valve. Based on the similar mathematical model, the natural frequencies are computed to confirm the dynamic characteristics of the valve and to determine the method for the stress analysis. Where more than one material exists, the pressure temperature rating and the standard calculation pressures are chosen for the weakest material in the stress calculation. These stresses are then compared with allowable stresses for the weakest material in the temperature range of interest.

Functional operability of safety-related active valves is assured by showing that the boundary joints, yokes, and similar structures have not failed, actuators do not freeze or bind, and structural integrity of the valve internals is not degraded. Valve end loads are provided in the valve design specification for safety-related active valves. External piping loads are kept within these specified limits. The faulted condition nozzle loads are considered in one of the following ways: (1) loads equivalent to the faulted condition nozzle loads are simultaneously applied specifications require the vendor to demonstrate the operability of the active valve when subjected to the loading combinations given in Tables 3.9-2.3 and 3.9-2.3A. In addition, the stresses are limited to the values given in Tables 3.9-5 and 3.9-6A.

Environmental service conditions for normal, abnormal and accident conditions are identified in Section 3.11. Safety-related active valves are qualified for operability during conditions where their operation is essential.

The valve specification requires that active valves be stroked during dynamic or static testing. For line-mounted valves, enveloping acceleration values from piping analysis are specified as required Input Motion (RIM). Valves of 3g for each of the two horizontal directions and 2g for the vertical direction are specified unless lower values are justified. For floor or wall mounted valves, required response spectra (RRS) are specified. The seismic accelerations in the three orthogonal directions are assumed to act simultaneously. Acceptance criteria is provided for structural failure, permanent deformation, performance characteristics, seat leakage, and malfunction of any appurtenances.

The qualification of valve body and extended structure as an assembly is performed by dynamic testing or static operability test supplemented by analysis.

Valves that are safety-related but can be classified as not having an extended structure, such as check, safety and relief valves are considered separately. These valves are characteristically simple in design, and their operation will not be affected by seismic accelerations or the maximum applied nozzle loads. The valve designs are compact, and there are no extended structures or masses whose motion could cause distortions that 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 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 by test and/or stress analysis of critical parts which may affect operability, including the faulted condition loads, the ability of the valve to operate is assured by the design features. Check valves also undergo the following: (1) in-shop hydrostatic test, (2) in-shop seat leakage test, (3) periodic in situvalve exercising, testing, and inspection to assure functional ability of the valve.

Active butterfly valves which are installed in piping by bolting between pipe flanges and having a cylindrical cross section of such proportions that the length of the valve parallel to the pipe run is equal to or less than the inside diameter of the valve, are exempted from the maximum applied end load qualification test. This is in accordance with ANSI B16.41, Annex D, Section D7.

The above methods provide assurance that safety-related active valves are qualified for operability during conditions where their operation is required.

3.9.3.3 Design and Installation Details for Mounting of Pressure-Relief Devices.

3.9.3.3.1 Design and Installation details for Mounting of Pressure-Relief Devices (NSSS Scope): Safety valves and relief valves are analyzed in accordance with the ASME Section III Code.

The method of analysis for safety valves and relief valves suitably accounts for the time-history of loads acting during and subsequent to valve opening; i.e., less than 1 second. The fluid-induced forcing functions are calculated for each safety valve and relief valve using one-dimensional equations for the conservation of mass, momentum, and energy.

The calculated forcing functions are applied at locations along the associated piping where a change in fluid flow direction occurs. Application of these forcing functions to the associated piping model constitutes the dynamic time-history analysis.

The dynamic response of the piping system is determined for the input forcing functions; therefore, a dynamic amplification factor is inherently accounted for in the analyses.

Snubbers or strut-type restraints are used as required. The stresses resulting from the loads produced by the sudden opening of a relief or safety valve are combined with stresses due to other pertinent loads and are shown to be within allowable limits of the ASME Section III Code. Also, the analyses shows that the loads applied to the nozzles of the safety and relief valves do not exceed the maximum loads specified by the manufacturer.

3.9.3.3.1.1 Pressurizer Safety and Relief System – The pressurizer safety and relief valve discharge piping systems provide overpressure protection for the RCS. The three spring-loaded safety valves, located on top of the pressurizer, are designed to prevent system pressure from exceeding design pressure by more than 10 percent. The two power-operated relief valves, also located on top of the pressurizer, are designed to prevent system pressure from exceeding the normal operating pressure by more than 100 psi. A water seal is maintained upstream of each valve prevents any leakage of hydrogen gas or steam through the valves. The valve outlet side is sloped to prevent the formulation of additional water pockets.

The pressurizer safety valves, manufactured by Crosby, are self-actuated spring loaded valves with backpressure compensation. The power-operated relief valves, manufactured by Garrett Air Research are solenoid actuated globe valves, capable of automatic operation via high pressure signal or remote manual operation. The safety valves and relief valves are located in the pressurizer cubicle and are supported by the attached piping which, in turn, is supported by a system of beams, struts, and snubbers.

If the pressure exceeds the setpoint and the valves open, the water slug from the loop seal discharges. The water slug, driven by high system pressure, generates transient thrust forces at each location where a change in flow direction occurs. The valve discharge conditions considered in the analysis of the Pressurizer Safety and Relief Valves (PSARV) piping systems are as follows: 1) the three safety valves are assumed to open simultaneously while the relief valves remain closed, and 2) the two relief valves open simultaneously while the safety vales are closed. In addition to these two cases, which consider water seal discharge (water slug followed by steam), solid water from the pressurizer (cold overpressure) is also investigated.

For each pressurizer safety and relief piping system , an analytical hydraulic model is developed. The piping from the pressurizer nozzle to the relief tank nozzle is modeled as a series of single pipes. The pressurizer is modeled as reservoir which contains steam at constant pressure (approximately 2,500 psia for safety system and approximately 2,350 psia for relief system) and at approximately 680°F. The pressurizer relief tank is modeled as a sink which contains steam and water mixture.

Fluid acceleration inside the pipe generates reaction forces on all segments of the line which are bounded at either end by an elbow or bend. Reaction forces resulting from fluid pressure and momentum variations are calculated. These forces are defined in terms of the fluid properties for the transient hydraulic analysis.

Unbalanced forces are calculated for each straight segment of pipe from the pressurizer to the relief tank. The time histories of these forces are used for the subsequent structural analysis of the pressurizer safety and relief lines.

The structural model used in the seismic analysis of the safety and relief lines is modified for the valve thrust analysis to represent the safety and relief valve discharge. The time-history hydraulic forces are applied to the piping system lump mass points. The dynamic solution for the valve thrust is obtained by using a modified predictor-corrector-integration technique and normal mode theory.

The time-history solution is performed in subprogram FIXFM3. The input to this subprogram consists of the natural frequencies and normal modes, applied forces, and nonlinear elements. The natural frequencies and normal modes for the modified pressurizer safety and relief line dynamic model are determined with the WESTDYN program. The support loads are computed by multiplying the support stiffness matrix ad the displacement vector at each support point. The time-history displacements of the FIXFM3 subprogram are used as input to the WESDYN2 subprogram to determine the internal forces, deflections, and stresses at each end of the piping elements.

The loading combinations considered in the analysis of the PSARV piping are given in table 3.9-2.4A. These load combinations are consistent with the final recommendations f the piping subcommittee of the Electric Power Research Institute (EPRI) pressurized water reactor (PWR) PSARV performance test program.

Pressure-relieving devices have been constructed, located, and installed so that they are readily accessible for inspection and repair and so that they cannot be readily rendered inoperative. Safety or relief valves have been set to relieve at a pressure not exceeding the design pressure of the vessel at the design temperature.

3.9.3.3.2 Design and Installation Details for Mounting of Pressure Relief Devices (BOP Scope): Pressure vessels have been protected by pressure relieving devices to meet applicable code requirements such as ASME Code, Section III, Section Viii, ANSI B31.1, RG 1.67 of October 1973.

The load due to reaction force from the opening and subsequent venting of a safety valve or relief valve(s) includes consideration of both momentum and pressure effects and has been computed either by dynamic time history analysis or by the static load method. The following formula has been used to calculate the reaction force:

$$F = \frac{W}{g} V + P A$$

where:

F = Reaction force, lb-force

W = Mass flow rate (relieving capacity stamped on the valve x 1.11), lb mass/sec

g = Gravitational constant, 32.2 lb-mass ft/lb-force sec^2

V = Exit velocity, ft/sec

P = Static gauge pressure at exit, lb-force/$in.^2$

A = Exit flow area, $in.^2$

When the reaction force F is calculated by the static load method, a dynamic load factor based on the relief/safety vale opening time and system dynamic characteristics is applied to the forces and moments due to the reaction force. Methods of analysis explained in Code Case 1569 (March 1973) of ASME Code, Section III, are followed, except for the discharge piping.

Fabrication and installation of the valve inlet nozzle to the header are in full compliance with the applicable provisions of ASME Code, Section III, Classes 2 and 3, for branch connections. Stresses in these pipes, including the effects of valve discharge thrust, are maintained within code limits.

Pressure-relieving devices have been constructed, located and installed so that they are readily accessible for inspection and repair and so that they cannot be readily rendered inoperative. Safety or relief valves have been set to relieve at a pressure not exceeding the design pressure of the vessel at the design temperature.

3.9.3.4 Component Supports.

3.9.3.4.1 NSSS Vendor-Supplied Component Supports: Component supports are designed in accordance with the following:

1. Linear Type Supports

   a. Normal – The allowable stresses of Appendix XVII of ASME B&PV Code Section III, as referenced in Subsection NF, are used for normal condition limits.

   b. Upset – Stress limits for upset conditions are the same as normal condition stress limits. This is consistent with Subsection NF of ASME B&PV Code Section III (NF-3230).

   c. Emergency – For emergency conditions, the allowable stresses or load ratings are 33 percent higher than those specified for normal conditions. This is consistent with Subsection NF of ASME B&PV Code Section III, in which (NF-3231) limits for emergency conditions are 33 percent greater than the normal condition limits.

   d. Faulted – Faulted condition limits are those specified in F-1370 of Appendix F of the ASME Code, Section III (NF-3230). The supports for active components are designed so that stresses are less than equal to S_y. Thus the operability of active components will not be endangered by the supports during faulted conditions.

2. Plate and Shell Type Supports

   a. Normal – Normal conditions limits are those specified in Subsection NF of ASME B&PV Code, Section III (NF-3220).

   b. Upset – Upset condition limits are those specified in Subsection NF of ASME B&PV Code, Section III (NF-3220).

   c. Emergency – For emergency conditions, the allowable stresses or load ratings are 20 percent higher than those specified for normal conditions.

   d. Faulted – Faulted condition limits are those specified in Subsection NF of ASME Code, Section III (NF-3220).

For active Class 2 or 3 pumps, support adequacy is proved by satisfying the criteria in Section 3.9.2.2. The requirements consist of both stress analysis and an evaluation of pump/motor support misalignment.

Active valves are, in general, supported only by the pipe attached to the body.

3. Hydraulic Snubbers

   The SG upper lateral support snubber design specification establishes requirements relating to maximum internal resistance to normal loadings, lockup velocity, stroke, material restrictions, spring rate, bleed rate, load capacity, hydraulic fluid properties, reservoir/tubing design, environmental conditions, design life, design analysis documentation, QA, NDE, functional testing, and shipping. The snubber manufacturer, Paul-Munroe Hydraulics, was required to perform functional tests demonstrating conformance to snubber performance requirements before, during, and after the application of normal, upset, and faulted loads. Elements of the test procedure are summarized below:

- Pressurize cylinder to faulted load, check seals.
- Cycle and bleed, check for binding.
- Plug valve port, check spring rate at faulted load.
- Install valves, check lockup velocity in each direction.
- Check bleed rate.
- Friction test, verify seals not damaged by above testing.

A discussion of the analytical models and techniques used in analyzing the reactor coolant loop piping, components, and supports is presented in Section 3.9.1.4. Load conditions/transients analyzed are normal operating and seismic (OBE), and SSE seismic combined with LOCA. The snubbers are modeled/analyzed using the STRUDL computer program. The average snubber stiffness is combined with the stiffness of the entire upper support and used in the seismic system analysis. Nonlinear tension and compression spring rates are used in the LOCA model.

The load ratios of actual/rated capacity are 0.73 for normal operating and OBE, and 0.54 for SSE seismic combined with LOCA.

3.9.3.4.2 Component Supports for BOP Components:

3.9.3.4.2.1 Applicable Codes and Standards – See Section 3.9.3.1.2.3.

3.9.3.4.2.2 Standard Components – Standard components are the manufacturer's catalog items for pipe supports. These have been designed according to ASME B&PV Code, Section III, Subsection NF.

3.9.3.4.2.3 Linear Type Supports – The design rules and stress limits which must be satisfied for the Design and Operating Conditions are given in NF-3230 (NF-3330).

a. Normal – The allowable stresses of appendix XVII of ASME B&PV Code, Section III, as referenced in Subsection NF, are used for normal condition limits.

b. Upset – Stress limits for upset conditions are the same as normal condition stress limits. This is consistent with Subsection NF of ASME B&PV Code, Section III (NF-3231.1[a]).

c. Emergency – For emergency conditions, the allowable stresses of load ratings are 33 percent higher than those specified for normal conditions. This is consistent with Subsection NF of ASME B&PV Code, Section III, in which (NF-3231.1[b]) limits for emergency conditions are 33 percent greater than the normal condition limits.

d. Faulted – For faulted condition, stress limits as referenced in (NF-3231.1[c]) of Subsection NF of the ASME B&PV Code are used.

3.9.3.4.2.4 Plate and Shell Type Supports – Plate and shell type supports are not used to support piping in the design of BOP systems. The design of plate and shell type supports for other components has been performed in accordance with ASME B&PV Section III Code.

3.9.3.4.2.5 General Design Considerations – The loadings, as specified in the design specifications, are taken into account in designing component supports for ASME code constructed items. These loadings include but are not limited to the following:

1. Weight of the component and normal contents under operating and test conditions

2. Weight of the component support

3. Superimposed loads and reactions induced by the adjacent system components

4. Dynamic loads, including loads caused by earthquake vibration

5. Restrained thermal expansion

6. Anchor and support movement effects

The combinations of loadings categorized with respect to plant operating conditions identified as Normal, Upset, Emergency, and Faulted which are specified for the design of supports for ASME Code constructed items are presented in Table 3.9-2.4. The stress limits for each plant operating condition are specified in Table 3.9-7, 3.9-7A, 3.9-7B, and 3.9-7C.

The recommendations of RG 1.124 applicable to the service limits and loading combinations for Class 1 linear supports are met as discussed in Table 3.9-2.4. Refer to Section 3.9.1.4.7, also.

Valves, in general are supported by the pipe attached to the valve. Exterior supports have been used both on the valve and also on the operator. Requirements for supports for pumps and vessels are included in the overall design and qualification of the component. Active pumps and valves are qualified for operability as described in Section 3.9.3.2 and any deformation of their supports is considered in the operability qualification.

3.9.3.4.2.6 Snubbers Used as Component Supports – The location and size of the snubbers are determined by stress analysis. The stress analysis is performed using the computer program mentioned in Section 3.9.1 and the loading combination given in Table 3.9-2.4. The location and line of action of a snubber are selected based on the necessity of limiting seismic stresses in the piping and nozzle loads on equipment. Snubbers are chosen in lieu of rigid supports where restricting thermal growth would induce excessive thermal stresses in the piping nozzle on loads or equipment. The snubbers are constructed to ASME B&PV Code, Section III, Subsection NF standards.

Snubbers are used I supporting piping and heavy operators of some valves. In the mathematical model, the snubbers are modeled as springs. The spring stiffness is the combined stiffness of the snubber and other supporting elements such as clamps, brackets, etc.

The design specification requires consideration of the following:

a. The mechanical snubber is considered a linear support. Design is in accordance with Subarticle NF-3200 Section III.

b. A certified stress report or certified load capacity data sheet is furnished showing the load capabilities of the snubber. Verification of the load carrying capability of the snubber is in accordance with NF-3132 of Section III.

c. The service loading of the snubber is equal to or less than the design strength established under Item b above for the particular loading condition.

d. The snubbers are designed for normal operation with a temperature range of 0° to 120°F and are capable of providing normal performance when exposed to an accident environmental temperature of 323°F for a minimum period of 8 hours.

e. The total movement during cyclic loading including lost motion and structural deflection, does not exceed ±0.06 in. at any load up to a rated load when subjected to cyclic loading in the frequency range of the 3 to 33 Hz.

f. When the snubber is subjected to a continuous load in either tension or compression, it shall continue to move without locking up.

g. The frictional resistance to normal movement is less than 2 percent of the rated load or 10 pounds, whichever is greater.

h. The snubbers meet the following requirements: For 400 lb through 5,000 lb load rated units, the minimum time required to travel one inch is 0.90 seconds at rated load; for 16,000 lb and greater load ratings, the minimum time required to travel 1 in. is 1.50 seconds at a rated load (as an alternative, the minimum time required to travel 1 in. is 3.0 seconds at 25 percent of the rated load); snubber backlash does not exceed 0.02 inches.

i. Dynamic Load Cycling – The unit will remain functional during and after being subjected to sinusoidal or step loadings at the rated load with frequencies between 3 Hz and 33 Hz at 3 Hz increments.

j. Low Temperature – The unit will remain functional after being subjected to a temperature of 0°F or colder for a minimum 8 hours.

k. High Temperature – The unit will remain functional after being subjected to a temperature of 323°F or higher for a minimum of 8 hours.

l. Life Test – The unit will remain functional after being stroked through a minimum of 80 percent of full travel for 40,000 cycles.

m. Faulted Load Static Test – The unit will remain functional after being subjected to the faulted load rating for 1 minute in tension and compression.

n. Side Load Static Test – The unit will remain functional after being subjected to a combination of the faulted load rating and a side load equivalent to 6g in the fully extended position.

o. For adjustable span snubbers, span is adjustable over a range of ±3-1/2 inches from the designed length without changing the operating position of the unit.

p. The design, procurement, manufacture, inspection, handling, testing, storage, and shipping of units and their component parts are performed in accordance with the Quality Assurance Program and the vendor's standard Quality Assurance procedures.

The design specification requires that an installation manual be provided by the manufacturer to ensure correct installation, including dimensional detailed drawings giving materials of construction with installation and adjustment instruction. Visual confirmation and inspection are required in the field. Also, the hot and cold position of the snubbers are measured during the preoperational testing stage.

Snubbers are located in order to most efficiently minimize stresses in the components and piping. Although there are no formal provisions for accessibility, access is provided for inspection, testing, repair, or replacement of snubbers by removing obstructions, if necessary.

All non-NSSS snubbers are of the mechanical type. The fabricator of the mechanical non-NSSS snubbers is the Anchor Darling Company.

Two types of tests are performed on the snubbers:

a. Production tests described in items f, g, and h (Section 3.9.3.4.2.6) are made on every unit.

b. Qualification tests described in items I, j, k, l, m and n (Section 3.9.3.4.2.6) are performed on randomly selected production models.

As only mechanical snubbers are used as component supports, there is no impact on the performance if the snubber by entrapped air or temperature on fluid properties.

Tabulations of snubbers utilized as supports for safety-related systems are not performed. This is in-line with Nuclear Regulatory Commission (NRC) Generic Letter 84-13.

If additional snubbers are installed after plant startup, documentation verifying operability and non-interference with normal plant operation will be established.

3.9.4 Control Rod Drive Systems

3.9.4.1 Descriptive Information on Control Rod Drive Systems.

3.9.4.1.1 Control Rod Drive Mechanism: CRDMs are located on the dome of the reactor vessel. They are coupled to RCCs which have absorber material over the entire length of the control rods and derive their name from this feature. The CRDMs are shown on Figure 3.9-5.

The primary function of the CRDM is to insert or withdraw RCCAs within the core to control reactivity and to shut down the reactor.

The CRDM is a magnetically operated jack. A magnetic jack is an arrangement of three electromagnets which are energized in a controlled sequence by a power cycler to insert or withdraw RCCSs in the reactor core in discrete steps. Rapid insertion of the RCCSs occurs when electrical power is interrupted.

The CRDM consists of four separate subassemblies. They are the pressure vessel, the coil stack assembly, the latch assembly, and the drive rod assembly.

1. For both Units 1 and 2, the pressure vessel originally included a latch housing and a rod travel housing connected by a threaded, seal-welded, maintenance joint, which facilitated replacement of the latch assembly. The closure at the top of the rod travel housing was a threaded plug with a canopy seal weld for pressure integrity. All of the original control rod drive mechanisms (CRDMs) for both Units have been replaced with a CRDM model that is similar in form, fit, and function to the original model of the CRDM except that the threaded joints between the CRDM pressure housings and the threaded joint between the CRDM assembly and the reactor vessel head penetration have been eliminated. The threaded and canopy seal welded joints featured on the original CRDMs have been replaced with full penetration butt welds.

   The latch housing is the lower portion of the vessel and contains the latch assembly and the auxiliary rod holdout device. 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. The coil stack assembly includes the coil housings, an electrical conduit and connector, and four operating coils: (a) the stationary gripper coil, (b) the moveable gripper coil, (c) the lift coil, and (d) the auxiliary rod holdout device 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 of the operating coils [2(a) through 2(c) in item 2 above] causes movement of the pole pieces and latches in the latch assembly which raises or lowers the drive rod assembly.

3. The latch assembly includes the guide tube, stationary pole pieces, moveable pole pieces, and two sets of latches: (a) the moveable gripper latch, and (b) the stationary gripper latch. The latches engage grooves in the drive rod assembly. The moveable gripper latches are moved up or down in 5/8-in. steps by the lift pole to raise or lower the drive rod. The stationary gripper latches hold the drive rod assembly while the moveable gripper latches are repositioned for the next 5/8-in. step.

4. The drive rod assembly includes a flexible coupling, a drive rod, a disconnect button, a disconnect rod, and a locking button.

   The drive rod has 5/8-in. grooves which receive the latches during holding or moving of the drive rod. The flexible coupling is attached to the drive rod and produces the means for coupling to the RCCA.

   The disconnect button, disconnect rod, and locking button provide positive locking of the coupling to the RCCS and permit remote disconnection of the drive rod.

The auxiliary rod holdout device (coil d in item 2) consists of an auxiliary latchbar at the bottom of the stationary gripper, a fourth operating coil at the bottom of the coil stack, and a latchbar return spring. The operating coil is completely isolated electrically from the mechanism latch assembly operating coils during normal plant operation. Energizing the auxiliary rod holdout device coil closes the rod holdout magnet gap, raising the latchbar into position behind the stationary gripper latch arms. When the stationary gripper latch coil is deenergized, the latch arms engage the auxiliary latchbar, preventing the release of the drive rod assembly.

The CRDM is a trip design. Tripping can occur during any part of the power cycler sequencing if power to the coils is interrupted.

The mechanism is capable of raising or lowering a 400-pound load (which includes the drive rod weight) at a rate of 45 in./min for withdrawal or 45 in./min for insertion. Withdrawal of the RCCA is accomplished by magnetic forces while insertion is accomplished 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 2,500 psia. The three operating coils are designed to operate at 392°F, with forced air cooling required to maintain that temperature during normal design operating conditions.

The auxiliary rod holdout device latch is self-locking with its load-carrying capability limited only by the design of the stationary gripper latch arms. These rod holdout devices are prevented from operation during plant operation by the following safety features:

1. The holdout device power supply is a constant voltage device set at approximately 290 vdc. The holdout device operating coil is sized to operate always at temperatures below 200°F and to limit operation above 500°F. The rod holdout operating coil resistance will rise due to the thermal coefficient of resistance of the winding and thus reduce ampere turn capability. The upper temperature limit is not absolute and is not a safety requirement but complements the electrical and control safety features.

2. The power supply for the rod holdout device is located outside the Containment. Provisions are made so that connection of the power supply to the rod holdout device coils cannot be made unless the Containment is entered and the connector cabinet is energized.

3. The power supply has one DC output line. A maximum of five rod holdout coils can be energized at a time. The input connections to the holdout coils are physically located so that each group of four of five coils must be individually connected to the power supply.

4. The flat-faced plunger magnet used to mechanically activate the holdout lock is a direct current device. An accidental short circuit to an AC source will not activate the device.

5. The only direct current in proximity to the auxiliary rod holdout device operating coil is the 150 V source for the mechanism operating coils. The auxiliary rod holdout coil is designed to provide sufficient ampere turns to operate the device only when voltage exceeds approximately 170 vdc and then without margin, when the coil is at a temperature of 20°C. A short circuit will not operate the holdout device.

The CRDMs shown schematically on Figure 3.9-5 withdraw and insert RCCAs as electrical pulses are received by the operator coils. An ON or OFF sequence, controlled by silicon rectifiers in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is measured by 48 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. The position of the shutdown rods is measured by 22 discrete coils.

During plant operation, the stationary gripper coil of the drive mechanism holds the RCCA in a static position until a stepping sequence is initiated, at which time the moveable gripper coil is energized.

3.9.4.1.2 Rod Cluster Control Assembly Withdrawal: The RCCA is withdrawn by repetition of the following sequence of event:

1. Moveable Gripper Coil (B) – ON

   The latch locking plunger raises and swings the moveable gripper latches into the drive rod assembly groove. A 1/16-in. 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 1/16 in. until the load of the drive rod assembly and attached control rod is transferred to the moveable 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-in. gap between the movable gripper pole and the lift pole closes and the drive rod assembly raises one step length (5/8 in.).

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, and 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) 1/16 inch. The 1/16-in vertical drive rod assembly movement transfers the drive rod assembly load from the moveable gripper latches to the stationary gripper latches.

5. Moveable Gripper Coil (B) – OFF

The latch locking plunger separates from the moveable gripper pole under the force of spring gravity. Three links, pinned to the plunger, swing the three moveable gripper latches out of the drive rod assembly groove.

6. Lift Coil (C) - OFF

The gap between the moveable gripper pole and lift pole opens. The moveable gripper latches drop 5/8 in. to a position adjacent to a drive rod assembly groove.

7. Repeat Step One

The sequence described above (1 through 6) is termed as one step or one cycle. The RCCA moves 5/8 in. 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-in. groove pitch) is raised 72 grooves per minute. The RCCA is thus withdrawn at a rate up to 45 in./min.

3.9.4.1.3 Rod Cluster Control Assembly Insertion: The sequence for RCCA 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-in. gap between the moveable gripper and lift pole closes. The moveable gripper latches are raised to a position adjacent to a drive rod assembly groove.

2. Moveable Gripper Coil (B) – ON

The latch locking plunger raises and swings the moveable gripper latches into a drive rod assembly groove. A 1/16-in. 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 RCCA, causes the stationary gripper latches and plunger to move downward 1/16 in. until the load of the drive rod assembly and attached RCCA is transferred to the moveable 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 separates the moveable 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) 1/16 inch. The 1/16-in. vertical drive rod assembly movement transfers the drive rod assembly load from the moveable gripper latches to the stationary gripper latches.

6. Moveable Gripper Coil (B) – OFF

The latch locking plunger separates from the moveable gripper pole under the force of a spring and gravity. Three links, pinned to the plunger, swing the three moveable gripper latches out of the drive rod assembly groove.

7. Repeat Step One

The sequence is repeated, as for RCCA withdrawal, up to 72 times per minute, which gives an insertion rate of 45 in./min.

3.9.4.1.4 Holding and Tripping of the Control Rods: During most of the plant operating time, the CRDMs hold the RCCAs in a withdrawn position 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 RCCAs hang suspended from the three latches.

If power to the stationary gripper coil is cut off, the combined weight of the drive rod assembly and the RCCA is sufficient to move latches out of the drive rod assembly groove. The control rod falls by gravity into the core. The trip occurs as the magnetic field, holding the stationary gripper plunger half against the stationary gripper pole, collapses and the stationary gripper plunger half is forced down by the weight acting upon the latches. After the RCCA is released by the mechanism, it falls freely until the control rods enter the dashpot section of their thimble tubes.

3.9.4.1.5 Withdrawal of Control Rods for Rapid Refueling: Following reactor cooldown and boration of the RCS for refueling, control rods are normally withdrawn by bank to their full-out position (actual rod withdrawal strategy is discussed in Section 9.1.4.2.2). With the control rods in this position, an operator manually activates the current limited rod holdout device auxiliary coil power supply. Each electrical group of control rods is individually patched into the power supply to close the magnet gap, raising the latchbar to the elevation of the stationary gripper latch arms. Power is turned off to the stationary gripper latch coil, latching the chosen rod withdrawn for refueling. Turing power off to the stationary gripper results in a check of the rod holdout device latch. The latched rod can now be removed from the auxiliary coil power supply and another group of rods locked out using the above sequence.

3.9.4.1.6 Insertion of Control Rods Following Rapid Refueling: Upon completion of the reactor refueling and following replacement of the upper package, the control rods may be inserted by an operator. The operator, after assuring that no rods are patched into the rod holdout power supply and that the power supply is completely disconnected and stored, inserts the rods by activating the moveable and stationary gripper latch coils.

3.9.4.1.7 Control Rod Position During Non-Rapid Refueling: During a non-rapid refueling, the control rods will normally remain in the fuel assemblies except during change-out operations or inspections (actual rod withdrawal strategy is discussed in Section 9.1.4.2.2).

3.9.4.2 Applicable CRDS Design Specifications. For those components in the Control Rod Drive System (CRDS) comprising portions of the RCPB, conformance with General Design Criteria (GDC) 15, 30, 31, 32 and 10CFR50, Section 50.55a is discussed in Sections 3.1 and 5.2. Conformance with RGs pertaining to materials suitability is described in Section 4.5 and 5.2.3.

3.9.4.2.1 Design Bases: Bases for temperature, stress on structural members, and material compatibility are imposed on the design of the reactivity control components.

3.9.4.2.2 Design Stresses: The CRDS is designed to withstand stresses originating from various operating conditions, as summarized in Table 5.2-1.

3.9.4.2.3 Allowable Stresses: For normal operating conditions, ASME B&PV Code, Section III is used. All RCPB components are analyzed as Class 1 components under Article NB-3000.

3.9.4.2.4 Dynamic Analysis: The cyclic stresses due to dynamic loads and deflections are combined with the 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 of the CRDS.

3.9.4.2.5 Control Rod Drive Mechanisms: The CRDMs pressure housings are Class 1 components designed to meet the stress requirements for normal operating conditions of ASME B&PV Code, Section III. 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 B&PV Code, Section III allowable stresses and to confirm its ability to trip when subjected to the seismic disturbance.

3.9.4.2.6 Control Rod Drive Mechanism Operational Requirements: The basic operational requirements for the CRDMs are:

1. 5/8-in. step

2. 168.125-in. travel

3. 400-pound maximum load

4. Step in at 45 in./min.; step out at 45 in./min.

5. Electrical power interruption shall initiate release of drive rod assembly during operation of the reactor

6. Trip delay time of less than 150 milliseconds free-fall of drive rod assembly shall be gin 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 with a maximum drive rod load of 200 pounds

7. 40-year design life with normal refurbishment; i.e., no less than 2.5×10^6 steps

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations.

3.9.4.3.1. 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 5.2-1

10. Pump overspeed

11. Seismic loads (OBE and SSE)

12. Blowdown forces (due to cold and hot leg branch pipe breaks)

The main objective of the analysis is to satisfy allowable stress limits, given in NB-3200 and NA Appendix F, 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 to limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Standard methods of deflections of these components. The dynamic behavior of the reactivity control components has been studied using experimental test data and experience from operating reactors.

3.9.4.3.2 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 RCCA. This always results in a reactivity decrease for control rods.

3.9.4.3.3 Latch Assembly and Coil Stack Assembly: With respect to the CRDM system as a whole, critical clearance are present in the following areas:

1. Latch assembly – diametral clearances

2. Latch arm – drive rod clearances

3. Coil stack assembly – thermal clearances

4. Coil fit in coil housing

The following defines clearances that are designed to provide reliable operation in the CRDM in these four critical areas. These clearances have been proven by life tests and actual field performance at operating plants.

1. 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 clearance of these parts at 68°F is 0.011 inches. At the maximum design temperature of 650°F, minimum clearance is 0.0045 in. At the maximum expected operating temperature of 550°F is 0.0057 inches.

2. Latch Arm – Drive Rod Clearances

The CRDM incorporates a load transfer action. The moveable or stationary gripper latches are not under load during engagement, as previously explained, due to load transfer action.

Figure 3.9-9 shows latch clearance variation with the drive rod as a result of minimum and maximum temperatures. Figure 3.9-10 shows clearance variations over the design temperature range.

3. Coil Stack Assembly – Thermal Clearances

The assembly clearance of the coil stack assembly over the latch housing was 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. 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 ID of the coil stack being 7.310 in. at 222°F and the maximum latch housing diameter being 7.302 in. 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.

Four coil stack assemblies were removed from four hot CRDMs mounted on 11.035-in. centers on a 550°F test loop, allowed to cool, and then placed without incident as a test to prove the preceding.

4. Coil Fit in Coil Housing

CRDM and coil housing clearances are selected so that coil heatup results in a close to tight fit. This is done to facilitate thermal transfer and coil cooling in a hot CRDM.

3.9.4.4 CRDM Performance Assurance Program (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 B&PV Code, Section III.

Internal components subjected to wear will withstand a minimum of three million steps without refurbishment, as confirmed by life tests (Ref. 3.9-10). Latch assembly inspection is recommended after 2.5 x 10^6 steps have been accumulated on a single CRDM.

To confirm the mechanical adequacy of the fuel assembly, the CRDM, and the RCCA, functional test programs are conducted on full-scale equipment. The prototype assembly is tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1,00 hours. Approximately 3 million steps and 600 trips are accumulated during testing. At the end of the test, the CRDM must operate satisfactorily.

The replacement CRDM lead latch assembly was tested cold and hot (>558°F) for 400 excursions each. No trips were performed. Verification that the trip time achieved by the replacement CRDMs meets the design requirement of 2.8 seconds from start of RCCA motion to dashpot entry was confirmed for each CRDM following replacement.

There are no significant differences between the prototype CRDMs and the production units. Design, materials, tolerances, and fabrication techniques are the same.

These tests have been reported in Reference 3.9-10.

If an RCCA cannot be moved by its mechanism, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip. Thus, inability to move one RCCA can be tolerated. More than one inoperable RCCA could be tolerated, but would impose additional demands on the plant operator. Therefore, the number of inoperable RCCAs has been limited to one as will be discussed by the Technical Specifications.

In order to demonstrate proper operation of the CRDM, RCCA partial-movement checks are performed on RCCAs. In addition, periodic drop tests of the RCCAs are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements, to verify that all rod hold out devices have been disengaged, to ensure core subcriticality after reactor trip, and to limit potential reactivity insertions from a hypothetical RCCA ejection. During these tests, the maximum acceptable drop time of each assembly is not greater than 2.8 seconds, at full flow and operating temperature, from the beginning of motion to dashpot entry.

Actual experience in operating many Westinghouse plants indicates excellent performance of CRDMs.

All units are production tested prior to shipment to confirm ability of the CRDM to meet design specification operational 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 to moveable gripper or moveable gripper to stationary gripper)	Step Length 5/8 ± 0.015 in. axial movement Load Transfer 0.047 in. nominal axial movement
Cold (ambient) performance test at design load – five full travel excursions	Operating Speed 45 in./min withdrawal; 45 in./min insertion Trip Delay Free-fall of drive rod to begin within 150 milliseconds

3.9.5 Reactor Pressure Vessel Internals

3.9.5.1 Design Arrangements. The RPV internals are described in the following paragraphs.

The components of the RPV internals are divided into three parts, consisting of the lower core support structure (including the entire core barrel and neutron 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 CRDMs, direct coolant flow past the fuel elements, direct coolant flow to the RPV head, provide gamma and neutron shielding, and guide 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. The lower core support is sized to provide the desired inlet flow distribution to the core. After passing through the core, the coolant enters the outlet plenum 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.

All the major material for the reactor internals is type 304 stainless steel. Parts not fabricated from type 304 stainless steel include bolts and dowel pins, which are fabricated from type 316 stainless steel, and radial support key bolts, which are fabricated of Inconel-750. These materials are listed in Table 5.2-10. There are no other materials used in the reactor internals or core support structures which are not otherwise include in ASME B&PV Code, Section III, Appendix I.

The discussions provided in Section 5.2.3.4 verify conformance of reactor internals and core support structures with RG 1.44, “Control of the Use of Sensitized Stainless Steel.”

The discussions provided in Section 5.2.3 are applicable to the welding of reactor internals and core support components.

The discussion provided in Section 5.2.3.4.6 verifies conformance of reactor internals and core support structures with RG 1.31, “Control of Stainless Steel Welding.”

The discussion provided in Section 5.2.3 verifies conformance of reactor internals and core support structures with RG 1.71, “Welder Qualification for Areas of Limited Accessibility.”

The only stainless steel materials used in the reactor core support structures which have yield strengths greater than 90,000 pounds are the 403 series used for holddown springs. The use o these materials is compatible with the reactor coolant and is acceptable based on the 1974 ASME B&PV Code, Case Number 1337.

All reactor internals are removable from the vessel for the purpose of inspection as well as for the inspection of the vessel internal surface.

Lower Core Support Assembly

The major structure and support member of the reactor internals is the lower core support assembly, shown on Figure 3.9-1. This support structure assembly consists of the core barrel, the core baffle, 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 RPV head flange, and its lower end is restrained in its transverse movement by a radial support system. Within the core barrel is an axial baffle assembly which is attached to the core barrel wall and forms the enclosure periphery of the assembled core. The lower core support, and principally the core barrel, serve to provided passageways and control for the coolant flow.

The lower core support is a member through which the necessary flow distribution holes for the fuel assemblies are machined. Adequate coolant distribution if obtained through the use of the lower core support.

The neutron shielding 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 in. wide by 147 in. long by 2.8 in. 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 guide by a preloaded spring device at the top and bottom to prevent sample movement. Additional details of the neutron shielding pads and irradiation specimen holders are given in Reference 3.9-12.

Vertically downward loads from weight, fuel assembly perload, control rod dynamic loading, hydraulic loads, and earthquake acceleration are carried 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 to be distributed by the lower radial support to the vessel wall, and to the core barrel flange. Transverse acceleration of the fuel assemblies is transmitted to the core barrel shell by direct connection of the lower core support the barrel wall and by a radial support-type connection of the upper core plate to slab-sided pins pressed into the core barrel. The main radial support system of the core barrel is accomplished by “key” and “keyway” joints to the RPV wall. At equally spaced points on the circumference, an Inconel block is welded to the vessel inner diameter. Another Inconel 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 farthest extremity and may be viewed as abeam 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 B&PV 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 cylindrical energy absorbers are contoured on their bottom surface to the RPV bottom head geometry. Their number and design are determined in order to limit, to less than yield, the stresses imposed on all components except the energy absorber. Assuming a downward vertical displacement, the potential energy of the system is absorbed mostly by the strain energy of the energy-absorbing devices.

Upper Core Support Assembly

The upper core support assembly, shown on Figures 3.9-2 and 3.9-3, consists of the upper 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 conduit. During refueling, when the upper internals and head arrangement to the storage sand. 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 guided 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 plate and guide tube.

The upper core support assembly, which is removed with the vessel head as a unit during refueling operation, 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 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 f the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the top of the fuel assemblies and engage the upper core plate as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the supper core support structure, the upper core support assembly, the fuel assemblies, and the control rods is 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 to the RPV 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 defection.

Incore Instrumentation Support Structures

The incore instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom (Figure 7.7-9 shows the Basic Flux-Mapping System).

The upper system utilizes the reactor head penetrations. Instrumentation port columns are slip-connected to in-line columns that are, in turn, fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouple conduits are supported from the columns of the upper core support system and are sealed, stainless steel tubes.

In addition to the upper incore instrumentation, there are RPV bottom port columns which carry the retractable, cold-worked stainless steel flux thimbles that are pushed upward into the reactor core. conduits extend from the bottom of the RPV down through the concrete shield area to a thimble seal line. The minimum bend radii are about 144 in., and the trailing ends of the thimbles (at the seal line) are extracted approximately 16 ft during refueling of the reactor to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the Containment atmosphere.

Mechanical seals between the retractable thimbles and conduits are provided at the seal line. During normal operation, the retractable thimbles are stationary and move only during refueling or for maintenance, at which time a space of approximately 16 ft above the seal line 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. RPV surveillance specimen capsules are covered in Section 5.3.1.6.

3.9.5.2 Design Loading Conditions. The design loading conditions that provide the basis for the design of the reactor internals are:

1. Fuel assembly weight

2. Fuel assembly spring forces

3. Internals weight

4. Control rod trip (equivalent static load)

5. Differential pressure

6. Spring preloads

7. Coolant flow forces (static)

8. Temperature gradients

9. Differences in thermal expansion

    a. Due to temperature differences

    b. Due to expansion of different materials

10. Interference between components

11. Vibration (mechanically or hydraulically induced)

12. One or more loops out of service

13. All operational transients listed in Table 3.9-8.

14. Pump overspeed

15. Seismic loads (OBE and SSE)

16 Blowdown forces (due to cold and hot leg branch pipe break)

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 also to 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 has been provide in Section 3.9.2.

As part of the evaluation of design loading conditions, extensive testing and inspection have been performed form 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 plant operation.

3.9.5.2.1 Normal and Upset: The normal and upset loading conditions that provide the basis for the design of the reactor internals are:

a) Fuel and reactor internals weight

b) Fuel and core component spring forces including spring preloading forces

c) Differential pressure and coolant flow forces

d) Temperature gradients

e) Vibratory loads including OBE seismic

f) The normal and upset operational thermal transients listed in 3.9.1.1.6 and 3.9.1.1.7

g) Control rod trip (equivalent static load)

h) Loads due to LOOP (s) out-of-serivce

i) Loss of load/pump overspeed

3.9.5.2.2 <u>Emergency Conditions</u>: The emergency loading conditions that provide the basis for the design of the reactor internals are:

a) Small LOCA

b) Small steam line break

c) Complete loss of flow

3.9.5.2.3 Faulted Conditions: The following faulted loading conditions are considered the most limiting and provide the basis for the design of the reactor internals are:

a) Rupture of an RCL branch pipe

b) SSE

3.9.5.3 Design Loading Categories. The combination of design loadings fits into either the normal, upset, or faulted condition as defined in the ASME B&PV Code, Section III. The allowable stress limits indicated in Subsections NG-3222 (Normal Conditions), NG-3223 (Upset Conditions), NG-3224 (Emergency Conditions) and Appendix F (Rules for Evaluating Faulted Conditions) are met.

Internal Structures are analyzed to meet the intent of the ASME Code in accordance with Subsection NG, paragraph NG-331 (c). Stresses in the Core Support Structure induced by interaction with internal structures are analyzed and shown to be in conformance with Core Support Limits. Design and construction for Core Support Structures meet Subsection NG in full.

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 components weights, hydraulic forces, and thermal gradients for the determination of the total stresses of the internals.

The RPV internals are designed to withstand stresses originating from various operating conditions as summarized in Table 3.9-8.

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.

Allowable Deflections

For normal conditions, downward vertical deflection of the lower core support plate is negligible. For the LOCA plus the design basis earthquake (DBE) condition, the defection criteria of critical internal structures are the limiting values given in Table 3.9-9. The corresponding no-loss-of-function limits are included in Table 3.9-9 for comparison purposes with the allowed criteria.

The criteria for the core drop accident are 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 RPV lower head in the hot condition is approximately 1 / 2 inch. An additional displacement of approximately 3 / 4 in. would occur due to stain of the energy-absorbing devices of the secondary core support; thus, the total drop distance is about 1-1/4 in., which is insufficient to permit the trips of the RCCA to come out of the guide thimble in the fuel assemblies.

Specifically, the secondary core support is a device which will not be used except during a hypothetical accident of the core support (core barrel, barrel flange, etc.). There are four supports in each reactor. This device limits the fall of the core and absorbs 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 he controlled volume of stainless steel, 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.

For additional information on design loading categories, see Section 3.9.1.

3.9.5.4 Design Bases. The design bases for the mechanical design of the RPV internals components are as follows:

1. The reactor internals, in conjunction with the fuel assemblies, shall 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 coolant for the RPV head shall be provided so that the temperature differences between the vessel flange and head do not result in leakage from the flange during reactor operation.

2. In addition to neutron shielding provided by the reactor coolant, a separate neutron pad assembly is provided to limit the exposure of the RPV in order to maintain the required ductility of the material for all modes of operation.

3. Provisions shall be made for installing incore instrumentation useful for the plant operation and vessel material test specimens required for a pressure vessel irradiation surveillance program.

4. The core internals are designed to withstand mechanical loads arising from OBE, SSE, and pipe ruptures and to meet the requirements of item 5, below.

5. The reactor shall have 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.

6. Following the DBA, the plant shall be capable of being shut down and cooled in an orderly fashion so that the 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.

7. The upper core support is designed to provide support to the upper package (i.e., vessel closure head, control rods and drive mechanisms, vessel closure studs, missile shield and CRDM cooling shroud) and provide radial protection to control rods during refueling.

The functional limitations for the core structures during the DBA are shown in Table 3.9-9. To ensure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited not to exceed the value shown in Table 3.9-9.

Details of the dynamic analyses, input forcing functions, and response loadings are presented in Section 3.9.2.

The basis for the design stress is identified below.

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. For materials not covered by the code, allowable stresses are established in the same manner as used in the code for materials of similar properties. It should be noted that the allowable stresses in Section III of the ASME Code are based on nonirradiated 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 DBA used for the core support structures are based on the 1974 edition of the ASME Code for Core support Structures, Subsection NG, and the criteria for faulted conditions.

3.9.6 Inservice Testing of Pumps and Valves

A program of preservice and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME B&PV Code and all applicable addenda as required by Paragraph (g) in Section 50.55a of 10CFR50, except for those exempted in accordance with UFSAR section 13.7.3. This testing program assures that Class 1, 2, and 3 components (pumps and valves) at the STPEGS will be in a state of operational readiness to perform their safety functions throughout the lifetime of the plant. Inservice tests conducted during the initial 120 month inspection interval shall comply (unless otherwise exempted) with the requirements in 10CFR50.55a (b) 12 months prior to the date of issuance of the operating license (subjected to the limitations and modifications listed in 10CFR50.55a (b)). Successive inspection intervals shall comply (where practical) with the requirements of the latest edition and addenda of the Code incorporated by reference in 10CFR50.55a (b) 12 months prior to the start of each 120 month inspection interval (subject to the limitations and modifications listed in 10CFR50.55a(b)). The testing schedules for applicable pumps and valves are included in the Pump and Valve Inservice Test Plan and the Technical Specifications.

3.9.6.1 Inservice Testing of Pumps. Class 1, 2, or 3 pumps specified in the Pump and Valve Inservice Test Plan and provided with an emergency power source shall be inservice tested (as applicable) according to the requirements of ASME B&PV Code Section XI, Subsection IWP and as modified by approved relief requests, except for those exempted in accordance with UFSAR section 13.7.3. The hydraulic and mechanical parameters to be measured or observed are also defined in

Subsection IWP. Measurement of parameters and appropriate records management will be executed in accordance with Articles IWP-4000 and IWP-6000 of Section XI, respectively.

The inservice test shall be conducted with constant speed pumps operating at nominal motor nameplate speed and with variable speed pumps adjusted to the reference speed. The resistance of the system shall be varied until either the measured differential pressure or the measured flow rate equals the corresponding reference value established in accordance with Article IWP-3000 of Section XI of the ASME B&PV Code. The test quantities shall then be measured or observed and recorded per Subsection IWP. Each measured test quantity shall then be compared with the reference value of the quantity. Any deviations determined shall be compared with the limits given in Subsection IWP, and corrective action shall be taken if necessary.

Reference values are defined as one or more fixed sets of values of the quantities measured or observed when the equipment is known to be operating acceptably. All subsequent test results shall be compared to these reference values or with new reference values established in accordance with Article IWP-3000 of Section XI of the ASME B&PV Code. Reference values shall be established based on the results of a test run during preoperational testing, initial operation or following maintenance activities that may have affected previous reference values. Reference values shall be at points of operation readily duplicated during subsequent inservice testing.

3.9.6.2 Inservice Testing of Valves. The valves specified in the Technical Specifications shall be tested according to the requirements of ASME B&PV Code Section XI, Subsection IWV and as modified by approved relief requests, except for those exempted in accordance with UFSAR section 13.7.3. Valves used for operating convenience only, such as manual vent, drain, instrument, and test valves, may be excluded from testing. Also excluded are external control protection systems responsible for sensing plant conditions and providing signals for valve operation.

Category A and B valves, as defined in Article IWV-2000 of Section XI of the B&PV Code, shall be exercised per Subsection IWV to the position required to fulfill their function, unless such operation is not practical. Category A valves will be tested for valve seat leakage in accordance with the Inservice Testing Program. Category C and D valves shall be tested according to the frequencies and procedures set forth in Articles IWV-3500 and IWV-3600 of the code. Appropriate records of these valve positions shall be kept.

When a valve or its control system has been replaced or repaired or has undergone maintenance that could affect its performance, it shall be tested before it is returned to service to demonstrate that the performance parameters which could be affected by the replacement, repair, or maintenance are within acceptable limits. Examples of maintenance that could affect valve performance parameters are: adjustment of stem packing; removal of the bonnet, stem assembly, or actuator; and disconnection of hydraulic or electrical lines.

REFERENCES

Section 3.9:

3.9-1 "Documentation of Selected Westinghouse Structural Analysis Computer Codes", WCAP-8252, Revision 1 (May 1977).

3.9-2 "Sample Analysis of a Class 1 Nuclear Piping System", , prepared by ASME Working Group on Piping, ASME Publication, 1972.

3.9-3 Bamford, W. H., and C. B. Buchalet, "Methods for Fracture Mechanics Analysis of Nuclear Reactor Vessels Under Severe Thermal Transients", WCAP-8510 (June 1976).

3.9-4 "Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests", WCAP-8317-A(March 1975).

3.9-5 "UHI Plant Internals Vibration Measurement Program and Pre and Post Hot Functional Examinations", WCAP-8517 (March 1975).

3.9-6 Trojan Final Safety Analysis Report, Appendix A-12, Docket No. 50-344.

3.9-7 Takeuchi, K., et al., "Multiflex – A Fortran-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics", WCAP-8708 and WCAP-8709 (Nonproprietary) (February 1976)

3.9-8 Fabic, S., "Computer Program WHAM for Calculation of Pressure Velocity and Force Transient in Liquid Filled Piping Networks", Daiser Engineers Report No. 67-49-R (November 1967).

3.9-9 Bohm, G. J., and J. P. LaFaille, "Reactor Internals Response Under a Blowdown Accident"" First International Conference on Structural Mechanics in Reactor Technology, Berlin (September 20-24, 1971).

3.9-10 Boyd, C. N., W. Ciaramitaro, and N. R. Singleton, "Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant", WCAP-8780 (May 1976).

3.9-11 Cooper, F. W., Jr., "17 x 17 Drive Line Components Tests – Phase 1B 11, 111, D-Loop-Drop and Deflection", WCAP-8446, Westinghouse Proprietary Class 2 and WCAP-8449, Westinghouse Nonproprietary (December 1974).

3.9-12 Kraus, S., "Neutron Shielding Pads", WCAP-7870 (May 1972).

REFERENCE (Continued)

Section 3.9:

3.9-13 Singleton, N. R., et al., “Four Loop PWR Internals Assurance and Test Program”, WCAP-7879, Proprietary (July 1972).

3.9-14 “SUPERPIPE – A Fortran-IV Computer Program for Structural Analysis and Design Checking for Piping Systems”, EDS Nuclear, CDC version (April 1977).

3.9-15 ADLPIPE – A digital computer program used for analysis of Complex Piping/Structural Systems, Arthur D. Little, Inc.

3.9-16 “Bench Mark Problem Solutions Employed for Verification of WECAN computer Program”, WCAP-8929 (June 1977).

3.9-17 Not Used.

3.9-18 NPS Piping Analysis Program User’s Manual – DYNAPO 4, NPS Incorporated

3.9-19 Altman, D.A., et al, “Verification of Upper Head Injection Reactor Vessel Internals by Preoperational Tests of the Sequoyah Power Plant”, WCAP-9646, March 1981.

3.9-20 K.F. Accornero, "Release of PIPESTRESS 3.4.08e", Westinghouse Letter MSE-SMT-96-245, dated December 18, 1996.

3.9-21 M.J. Malone, "WESPLAT on the SUN 4", Westinghouse Letter FDRT-SSD-003, dated January 4, 1996.

3.9-22 NEWKFAC verified via Westinghouse Calculation SAE / FSE-C-TGX-0182 (3/26/98)

TABLE 3.9-1.1

LIST OF ACTIVE COMPONENTS

Pump	System	System Designator	ANS Safety Class	Pump Tag Number
Containment Spray Pump 1, 2, 3	CSS	CS	2	2N101NPA101A,B,C
Boric Acid Transfer Pump 1, 2	CVCS	CV	3	3R171NPA103A,B
Centrifugal Charging Pump 1, 2	CVCS	CV	2	2R171NPA101A,B
High-Head Safety Injection Pump 1, 2, 3	SIS	SI	2	2N121NPA101A,B,C
Low-Head Safety Injection Pump 1, 2, 3	SIS	SI	2	2N121NPA102A,B
Spent Fuel Pool Pump 1, 2	SFPCCS	FC	3	3R211NPA101A,B
Auxiliary Feedwater Pump 1, 2, 3, 4	AFW	AF	3*	3S141MPA01,02,02,04
Component Cooling Water Pump 1, 2, 3	CCW	CC	3*	3R201NPA101A,B,C
Essential Cooling Water Pump 1, 2, 3	ECW	EW	3*	3R281NPA101A,B,C
Essential Cooling Water Screen Wash Pump 1, 2, 3	ECW	EW	3*	2R281NPA102A,B,C
Essential Cooling Water Traveling Water Screens 1, 2, 3	ECW	EW	3*	3R281NTW101A,B,C

* BOP scope of supply

TABLE 3.9-1.1 (Continued)

LIST OF ACTIVE COMPONENTS

Pump	System	System Designator	ANS Safety Class	Pump Tag Number
Reactor Makeup Water Pump 1, 2	RMWS	RM	3*	3R271NPA101A,B
Essential Chilled Water Pumps 1, 2, 3	Chilled Water	CH	3*	3V111VPA004,005,006
Essential Cooling Water Self Cleaning Strainers 1, 2, 3	ECW	EW	3*	3R281NSP101A,B,C
Residual Heat Removal Pump 1, 2, 3,	RHR	RH	2	2R161NPA101A 2R161NPA101B 2R161NPA101C

* BOP scope of supply

TABLE 3.9-1.2

ACTIVE VALVES (BOP SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
Component Cooling Water	CC0013	RHR Hx supply	16	Check	Process flow	2	2, 3
	CC0123	RHR Hx supply	16	Check	Process flow	2	2, 3
	CC0183	RHR Hx supply	16	Check	Process flow	2	2, 3
	CC0392	RCDT Hx supply	4	Gate	Motor	3	5
	CC0058	RCFC supply	14	Check	Process flow	2	2, 4
	CC0138	RCFC supply	14	Check	Process flow	2	2, 4
	CC0198	RCFC supply	14	Check	Process flow	2	2, 4
	CC0393	Excess letdown Hx supply	6	Butterfly	Motor	3	5
	CC0297	Excess letdown Hx & RCDT supply	6	Butterfly	Motor	3	5
	CC0768	Charging pumps supply	6	Butterfly	Motor	3	4
	CC0770	Charging pumps supply	6	Butterfly	Motor	3	4
	CC0771	Charging pumps supply	6	Butterfly	Motor	3	4
	CC0772	Charging pumps return	6	Butterfly	Motor	3	4
	CC0774	Charging pumps return	6	Butterfly	Motor	3	4
	CC0775	Charging pumps return	6	Butterfly	Motor	3	4
	CC0137	Chill water to RCFC supply isolation	8	Butterfly	Motor	2	2, 4, 5
	CC0149	Chill water to RCFC return isolation	8	Butterfly	Motor	2	2, 4, 5
	CC0199	Chill water to RCFC supply isolation	8	Butterfly	Motor	2	2, 4, 5
	CC0209	Chill water to RCFC return isolation	8	Butterfly	Motor	2	2, 4, 5
	CC0059	Chill water to RCFC supply isolation	8	Butterfly	Motor	2	2, 4, 5
	CC0070	Chill water to RCFC return isolation	8	Butterfly	Motor	2	2, 4, 5
	FV0862	Chill water to RCFC return isolation	8	Butterfly	Air	2	2, 4, 5
	FV0863	Chill water to RCFC return isolation	8	Butterfly	Air	2	2, 4, 5
	FV0864	Chill water to RCFC return isolation	8	Butterfly	Air	2	2, 4, 5

TABLE 3.9-1.2 (Continued)

ACTIVE VALVES (BOP SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
Component Cooling Water (Cont'd)	CC0051	Common Header return	24	Check	Process flow	3	4, 5
	CC0131	Common Header return	24	Check	Process flow	3	4, 5
	CC0191	Common Header return	24	Check	Process flow	3	4, 5
	CC0315	Common Header supply	24	Check	Process flow	3	4, 5
	CC0313	Common Header supply	24	Check	Process flow	3	4, 5
	CC0311	Common Header supply	24	Check	Process flow	3	4, 5
	CC0291	RCP supply	12	Butterfly	Motor	2	2
	CC0318	RCP supply	12	Butterfly	Motor	2	2
	CC0404	RCP return	12	Butterfly	Motor	2	2
	CC0403	RCP return	12	Butterfly	Motor	2	2
	CC0542	RCP return	12	Butterfly	Motor	2	2
	CC0764	Non-essential header return	18	Check	Process flow	3	5
	CC0765	Non-essential header return	18	Check	Process flow	3	5
	CC0068	RCFC return	14	Butterfly	Motor	2	2, 4
	CC0147	RCFC return	14	Butterfly	Motor	2	2, 4
	CC0208	RCFC return	14	Butterfly	Motor	2	2, 4
	CC0148	RCFC return	14	Butterfly	Motor	2	2, 4
	CC0197	RCFC supply	14	Butterfly	Motor	2	2, 4
	CC0210	RCFC return	14	Butterfly	Motor	2	2, 4
	CC0057	RCFC supply	14	Butterfly	Motor	2	2, 4
	CC0069	RCFC return	14	Butterfly	Motor	2	2, 4
	CC0136	RCFC supply	14	Butterfly	Motor	2	2, 4
	CC0319	RCP supply	12	Check	Process flow	2	2
	CC0012	RHR Hx supply	16	Butterfly	Motor	2	2
	CC0049	RHR Hx return	16	Butterfly	Motor	2	2
	CC0050	RHR Hx return	16	Butterfly	Motor	2	2
	CC0122	RHR Hx supply	16	Butterfly	Motor	2	2
	CC0129	RHR Hx return	16	Butterfly	Motor	2	2
	CC0130	RHR Hx return	16	Butterfly	Motor	2	2
	CC0182	RHR Hx supply	16	Butterfly	Motor	2	2
	CC0189	RHR Hx return	16	Butterfly	Motor	2	2
	CC0190	RHR Hx return	16	Butterfly	Motor	2	2
	CC0642	CCW Hx bypass	16	Butterfly	Motor	3	3, 4

TABLE 3.9-1.2 (Continued)

ACTIVE VALVES (BOP SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
Component Cooling Water (Cont'd)	CC0644	CCW Hx bypass	16	Butterfly	Motor	3	3, 4
	CC0646	CCW Hx bypass	16	Butterfly	Motor	3	3, 4
	CC0540	RCDT Hx return	4	Check	Process flow	3	5
	CC0541	RCDT Hx return	4	Check	Process flow	3	5
	CC0032	SFP Hx supply	18	Butterfly	Motor	3	5
	CC0236	Non-essential header supply	18	Butterfly	Motor	3	5
	CC0235	Non-essential header supply	18	Butterfly	Motor	3	5
	CC0447	SFP Hx supply	18	Butterfly	Motor	3	5
	CC0402	Excess letdown Hx return	6	Check	Process flow	3	5
	CC0763	Excess letdown Hx return	6	Check	Process flow	3	5
	CC0052	Common Header return	24	Butterfly	Motor	3	4, 5
	CC0132	Common Header return	24	Butterfly	Motor	3	4, 5
	CC0192	Common Header return	24	Butterfly	Motor	3	4, 5
	CC0312	Common Header supply	24	Butterfly	Motor	3	4, 5
	CC0314	Common Header supply	24	Butterfly	Motor	3	4, 5
	CC0316	Common Header supply	24	Butterfly	Motor	3	4, 5
	CC0643	CCW Hx discharge	24	Butterfly	Motor	3	3, 4
	CC0645	CCW Hx discharge	24	Butterfly	Motor	3	3, 4
	CC0647	CCW Hx discharge	24	Butterfly	Motor	3	3, 4
	CC0346	RCP Thermal Barrier supply	2	Check	Process flow	3	1
	CC0758	RCP Thermal Barrier supply	2	Check	Process flow	3	1
	CC0327	RCP Thermal Barrier supply	2	Check	Process flow	3	1
	CC0759	RCP Thermal Barrier supply	2	Check	Process flow	3	1
	CC0321	RCP Thermal Barrier supply	2	Check	Process flow	3	1
	CC0756	RCP Thermal Barrier supply	2	Check	Process flow	3	1
	CC0363	RCP Thermal Barrier supply	2	Check	Process flow	3	1
	CC0757	RCP Thermal Barrier supply	2	Check	Process flow	3	1
	FV-4657	Charging pumps return header	6	Butterfly	Air	3	5
	FV-4656	Charging pumps supply header	6	Butterfly	Air	3	5
	FV-4493	RCP return	12	Butterfly	Air	2	2

TABLE 3.9-1.2 (Continued)

ACTIVE VALVES (BOP SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
Component Cooling Water (Cont'd)	FV-4531	RHR Hx outlet	16	Butterfly	Air	3	3, 4
	FV-4565	RHR Hx outlet	16	Butterfly	Air	3	3, 4
	FV-4548	RHR Hx outlet	16	Butterfly	Air	3	3, 4
	FV-4540	PASS supply	1-1/2	Globe	Solenoid	3	5
	FV-4541	PASS supply	1-1/2	Globe	Solenoid	3	5
	FV-4620	RCP Thermal Barrier return	3	Globe	Process fluid	3	1
	FV-4621	RCP Thermal Barrier return	3	Globe	Process fluid	3	1
	FV-4626	RCP Thermal Barrier return	3	Globe	Process fluid	3	1
	FV-4627	RCP Thermal Barrier return	3	Globe	Process fluid	3	1
	FV-4632	RCP Thermal Barrier return	3	Globe	Process fluid	3	1
	FV-4633	RCP Thermal Barrier return	3	Globe	Process fluid	3	1
	FV-4638	RCP Thermal Barrier return	3	Globe	Process fluid	3	1
	FV-4639	RCP Thermal Barrier return	3	Globe	Process fluid	3	1
Liquid Waste Processing	FV-4920	RCDT vent	1	Globe	Solenoid	2	2
	FV-4913	RCDT discharge	3	Globe	Air	2	2
	WL0312	RCDT discharge	3	Gate	Motor	2	2
	FV-4919	RCDT vent	1	Globe	Air	2	2
Fire Protection	FP0756	RCB supply	6	Gate	Motor	2	2
	FP0943	RCB supply	6	Check	Process flow	2	2
Post Accident Sampling	FV-2454	Sample isolation	1	Globe	Solenoid	2	2
	FV-2453	Sample isolation	1	Globe	Solenoid	2	2
	FV-2455	Sample isolation	1	Globe	Solenoid	2	2
	FV-2456	Sample isolation	1	Globe	Solenoid	2	2
	FV-2458	Sample isolation	1	Globe	Solenoid	2	2
	FV-2457	Sample isolation	1	Globe	Solenoid	2	2
	FV-2455A	Sample isolation	1	Globe	Solenoid	2	2

TABLE 3.9-1.2 (Continued)

ACTIVE VALVES (BOP SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
Hydrogen Monitoring	FV-4100	Sample point selection	1	Globe	Solenoid	2	5
	FV-4124	Sample point selection	1	Globe	Solenoid	2	5
	FV-4125	Sample point selection	1	Globe	Solenoid	2	5
	FV-4126	Sample point selection	1	Globe	Solenoid	2	5
	FV-4101	RCB isolation	1	Globe	Solenoid	2	2,5
	FV-4127	RCB isolation	1	Globe	Solenoid	2	2, 5
	FV-4128	RCB isolation	1	Globe	Solenoid	2	2, 5
	FV-4103	Sample point selection	1	Globe	Solenoid	2	5
	FV-4129	Sample point selection	1	Globe	Solenoid	2	5
	FV-4130	Sample point selection	1	Globe	Solenoid	2	5
	FV-4131	Sample point selection	1	Globe	Solenoid	2	5
	FV-4104	RCB isolation	1	Globe	Solenoid	2	2, 5
	FV-4133	RCB isolation	1	Globe	Solenoid	2	2, 5
	FV-4134	RCB isolation	1	Globe	Solenoid	2	2, 5
	FV-4135	RCB isolation	1	Globe	Solenoid	2	2, 5
	FV-4136	RCB isolation	1	Globe	Solenoid	2	2, 5
Essential Cooling Water	EW0121	Pump discharge	30	Butterfly	Motor	3	3, 4
	EW0137	Pump discharge	30	Butterfly	Motor	3	3, 4
	EW0151	Pump discharge	30	Butterfly	Motor	3	3, 4
	FV-6914	Screen wash booster pump discharge	3	Globe	Air	3	5
	FV-6924	Screen wash booster pump discharge	3	Globe	Air	3	5
	FV-6934	Screen wash booster pump discharge	3	Globe	Air	3	5
	FV-6935	ECP blowdown	4	Globe	Air	3	5
	FV-6936	ECP blowdown	4	Globe	Air	3	5
	FV-6937	ECP blowdown	4	Globe	Air	3	5

TABLE 3.9-1.2 (Continued)

ACTIVE VALVES (BOP SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
Essential Cooling Water (Cont'd)	EW0006	Pump discharge	30	Check	Process flow	3	3, 4
	EW0042	Pump discharge	30	Check	Process flow	3	3, 4
	EW0079	Pump discharge	30	Check	Process flow	3	3, 4
	EW0403	Strainer Backwash	6	Check	Process flow	3	5
	EW0404	Strainer Backwash	6	Check	Process Flow	3	5
	EW0405	Strainer Backwash	6	Check	Process flow	3	5
	EW0370A	Pump column vent	3	Check	Process flow	3	5
	EW0370B	Pump column vent	3	Check	Process flow	3	5
	EW0370C	Pump column vent	3	Check	Process flow	3	5
	EW0253	Screen wash supply	3	Check	Process flow	3	5
	EW0254	Screen wash supply	3	Check	Process flow	3	5
	EW0255	Screen wash supply	3	Check	Process flow	3	5
Radioactive Equipment Floor Drain	ED0064	Containment sump isolation	3	Gate	Motor	2	2
	FV-7800	Containment sump isolation	3	Globe	Air	2	2
Auxiliary Feedwater	FV-7523	Flow control	4	Globe	Motor	3	3, 4
	FV-7524	Flow control	4	Globe	Motor	3	3, 4
	FV-7525	Flow control	4	Globe	Motor	3	3, 4
	FV-7526	Flow control	4	Globe	Motor	3	3, 4
	FV-7515	Cross-connect	4	Globe	Air	3	5
	FV-7516	Cross-connect	4	Globe	Air	3	5
	FV-7517	Cross-connect	4	Globe	Air	3	5
	FV-7518	Cross-connect	4	Globe	Air	3	5
	AF0019	Isolation	4	Stop Check	Motor	2	2, 3, 4
	AF0048	Isolation	4	Stop Check	Motor	2	2, 3, 4
	AF0065	Isolation	4	Stop Check	Motor	2	2, 3, 4
	AF0085	Isolation	4	Stop Check	Motor	2	2, 3, 4
	MS0143	Turbine steam inlet	4	Stop Check	Motor	2	2, 3
	FV-0143	Turbine steam inlet bypass	1	Globe	Solenoid	2	2, 3
	MS-0514	Turbine trip and throttle valve	4	Globe	Motor	3	3, 4

TABLE 3.9-1.2 (Continued)

ACTIVE VALVES (BOP SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
Auxiliary Feedwater (Cont'd)	AF0011	Pump discharge check/recirc	4	Auto Recirc. Check	Process flow	3	3, 4
	AF0036	Pump discharge check/recirc	4	Auto Recirc. Check	Process flow	3	3, 4
	AF0058	Pump discharge check/recirc	4	Auto Recirc. Check	Process flow	3	3, 4
	AF0091	Pump discharge check/recirc	4	Auto Recirc. Check	Process flow	3	3, 4
	AF0119	SG supply	8	Check	Process flow	2	2, 3, 4
	AF0120	SG supply	8	Check	Process flow	2	2, 3, 4
	AF0121	SG supply	8	Check	Process flow	2	2, 3, 4
	AF0122	SG supply	8	Check	Process flow	2	2, 3, 4
Main Steam	FSV-7414*	MSIV	30	Y-Pattern	Air	2	2
	FSV-7424*	MSIV	30	Y-Pattern	Air	2	2
	FSV-7434*	MSIV	30	Y-Pattern	Air	2	2
	FSV-7444*	MSIV	30	Y-Pattern	Air	2	2
	FV-7412	MSIV bypass	4	Globe	Air	2	2
	FV-7422	MSIV bypass	4	Globe	Air	2	2
	FV-7432	MSIV bypass	4	Globe	Air	2	2
	FV-7442	MSIV bypass	4	Globe	Air	2	2
	PSV-7410	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7410A	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7410B	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7410C	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7410D	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7420	Main Steam Safety Valve	6x10	Safety	High pressure	2	1

TABLE 3.9-1.2 (Continued)

ACTIVE VALVES (BOP SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
Main Steam (Cont'd)	PSV-7420A	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7420B	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7420C	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7420D	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7430	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7430A	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7430B	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7430C	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7430D	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7440	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7440A	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7440B	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7440C	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PSV-7440D	Main Steam Safety Valve	6x10	Safety	High pressure	2	1
	PV-7411	Steam Generator PORV	8x10	Drag	Electrohydraulic	2	1, 3, 4
	PV-7421	Steam Generator PORV	8x10	Drag	Electrohydraulic	2	1, 3, 4
	PV-7431	Steam Generator PORV	8x10	Drag	Electrohydraulic	2	1, 3, 4
	PV-7441	Steam Generator PORV	8x10	Drag	Electrohydraulic	2	1, 3, 4
Reactor Makeup Water	FV-7663	Non-Essential supply header	4	Globe	Air	3	5
	FV-7659	Non-Essential supply header	4	Globe	Air	3	5
	RM0003	Pump discharge	4	Check	Process flow	3	5
	RM0010	Pump discharge	4	Check	Process flow	3	5
	RM0013	Recycle supply to storage tank	2	Check	Process flow	3	5
Feedwater System	FV-7141	Feedwater Isolation	18	Gate	Hydraulic	2	2
	FV-7142	Feedwater Isolation	18	Gate	Hydraulic	2	2
	FV-7143	Feedwater Isolation	18	Gate	Hydraulic	2	2
	FV-7144	Feedwater Isolation	18	Gate	Hydraulic	2	2
	FW0062	Feedwater check	18	Check	Process flow	2	2
	FW0066	Feedwater check	18	Check	Process flow	2	2
	FW0067	Feedwater check	18	Check	Process flow	2	2
	FW0249	Feedwater check	18	Check	Process flow	2	2

TABLE 3.9-1.2 (Continued)

ACTIVE VALVES (BOP SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
Feedwater System (Cont'd)	FV-7145A	Feedwater isolation bypass	2	Globe	Air	2	2
	FV-7146A	Feedwater isolation bypass	2	Globe	Air	2	2
	FV-7147A	Feedwater isolation bypass	2	Globe	Air	2	2
	FV-7148A	Feedwater isolation bypass	2	Globe	Air	2	2
	FV-7189	SG feed preheater bypass	3	Globe	Air	2	2
	FV-7190	SG feed preheater bypass	3	Globe	Air	2	2
	FV-7191	SG feed preheater bypass	3	Globe	Air	2	2
	FV-7192	SG feed preheater bypass	3	Globe	Air	2	2
Sample System	FV-4461	RHR sample	1	Globe	Air	2	2
	FV-4456	RCS hot leg sample	1	Globe	Air	2	2
	FV-4452	Pressurizer vapor sample	1	Globe	Air	2	2
	FV-4466	SI accumulator sample	1	Globe	Air	2	2
	FV-4451B	Pressurizer liquid sample	1	Globe	Air	2	2
	FV-4450	Pressurizer vapor sample	1	Globe	Solenoid	2	2
	FV-4451	Pressurizer liquid sample	1	Globe	Solenoid	2	2
	FV-4454	RCS hot leg sample	1	Globe	Solenoid	2	2
	FV-4455	RCS hot leg sample	1	Globe	Solenoid	2	2
	FV-4458	RHR sample	1	Globe	Solenoid	2	5
	FV-4459	RHR sample	1	Globe	Solenoid	2	5
	FV-4460	RHR sample	1	Globe	Solenoid	2	5
	FV-4823	RHR sample	1	Globe	Solenoid	2	2
	FV-4824	SI accumulator sample	1	Globe	Solenoid	2	2
Instrument Air	FV-8565	Containment isolation	2	Ball	Air	2	2
	IA0541	Containment isolation	2	Check	Process flow	2	2
Steam Generator Blowdown	FV-4150	Blowdown line isolation	4	Globe	Air	2	2
	FV-4151	Blowdown line isolation	4	Globe	Air	2	2
	FV-4152	Blowdown line isolation	4	Globe	Air	2	2
	FV-4153	Blowdown line isolation	4	Globe	Air	2	2
	FV-4186	Sample line isolation	1	Y-Pattern	Solenoid	2	2
	FV-4187	Sample line isolation	1	Y-Pattern	Solenoid	2	2
	FV-4188	Sample line isolation	1	Y-Pattern	Solenoid	2	2
	FV-4189	Sample line isolation	1	Y-Pattern	Solenoid	2	2

TABLE 3.9-1.2 (Continued)

ACTIVE VALVES (BOP SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
Steam Generator Blowdown (Cont'd)	FV-4186A	Sample line isolation	1	Y-Pattern	Solenoid	2	2
	FV-4187A	Sample line isolation	1	Y-Pattern	Solenoid	2	2
	FV-4188A	Sample line isolation	1	Y-Pattern	Solenoid	2	2
	FV-4189A	Sample line isolation	1	Y-Pattern	Solenoid	2	2
Auxiliary Steam	FV-8838A	Steam line isolation	6	Globe	Air	3	5
	FV-8838B	Steam line isolation	6	Globe	Air	3	5
Personnel Airlock	FV-1025	Airlock air supply isolation	1/2	Globe	Solenoid	2	2
	FV-1026	Airlock air supply isolation	1/2	Globe	Solenoid	2	2
	FV-1027	Airlock auto leak rate monitoring isolation	1/2	Globe	Solenoid	2	2
	FV-1028	Airlock auto leak rate monitoring isolation	1/2	Globe	Solenoid	2	2
Containment Purge	FV-9776	Supplementary purge isolation	18	Butterfly	Air	2	2
	HC0003	Supplementary purge isolation	18	Butterfly	Motor	2	2
	HC0005	Supplementary purge isolation	18	Butterfly	Motor	2	2
	FV-9777	Supplementary purge isolation	18	Butterfly	Air	2	2
	HC0007	Normal purge isolation	48	Butterfly	Motor	2	5
	HC0008	Normal purge isolation	48	Butterfly	Motor	2	5
	HC0009	Normal purge isolation	48	Butterfly	Motor	2	5
	HC0010	Normal purge isolation	48	Butterfly	Motor	2	5
Radiation Monitoring	RA0001	RCB isolation	1	Ball	Motor	2	2
	RA0003	RCB isolation	1	Ball	Motor	2	2
	RA0004	RCB isolation	1	Ball	Motor	2	2
	RA0006	RCB isolation	1	Ball	Motor	2	2
Essential Chilled Water	TV-9476A	Control room AHU discharge	2	Butterfly	Air	3	5
	TV-9486A	Control room AHU discharge	2	Butterfly	Air	3	5
	TV-9496A	Control room AHU discharge	2	Butterfly	Air	3	5
	TV-9476B	Control room AHU bypass	2	Butterfly	Air	3	5
	TV-9486B	Control room AHU bypass	2	Butterfly	Air	3	5
	TV-9496B	Control room AHU bypass	2	Butterfly	Air	3	5
	TV-9477A	Main EAB AHU discharge	4	Butterfly	Air	3	5
	TV-9487A	Main EAB AHU discharge	4	Butterfly	Air	3	5
	TV-9497A	Main EAB AHU discharge	4	Butterfly	Air	3	5

TABLE 3.9-1.2 (Continued)

ACTIVE VALVES (BOP SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
Essential	TV-9477B	Main EAB AHU bypass	4	Butterfly	Air	3	5
Chilled Water	TV-9487B	Main EAB AHU bypass	4	Butterfly	Air	3	5
(Cont'd)	TV-9497B	Main EAB AHU bypass	4	Butterfly	Air	3	5
	CH0286	Pump Discharge	8	Check	Process flow	3	5
	CH0295	Pump Discharge	8	Check	Process flow	3	5
	CH0304	Pump Discharge	8	Check	Process flow	3	5
Diesel Generator	DO0056	Fuel Oil Return	1	Check	Process flow	3	5
Fuel Oil	DO0062	Fuel Oil Return	1	Check	Process flow	3	5
	DO0068	Fuel Oil Return	1	Check	Process flow	3	5
	DO0126	Fuel Oil Return	1	Check	Process flow	3	5
	DO0127	Fuel Oil Return	1	Check	Process flow	3	5
	DO0128	Fuel Oil Return	1	Check	Process flow	3	5

[a] Abbreviations

MSIV - Main Steam Isolation Valve
PORV - Power Operated Relief Valve
RCDT - Reactor Coolant Drain Tank
RHR - Residual Heat Removal
Hx - Heat Exchanger
RCFC - Reactor Containment Fan Cooler
RCP - Reactor Coolant Pump
NNS - Non Nuclear Safety
PASS - Post Accident Sampling System
SG - Steam Generator
AHU - Air Handling Unit
RCB - Reactor Containment Building
RCS - Reactor Coolant System

[b] Active Status

(1) Pressure Boundary Integrity
(2) Containment isolation - includes steam line isolation and feedwater isolation
(3) Emergency Core Cooling
(4) Cold Shutdown
(5) Auxiliary Process Support Function
* Containment isolation in modes 5 & 6

[c] Valve numbers followed by an asterisk "*" are supplied by Westinghouse. The remainder of the valves in this table are supplied by Bechtel.

TABLE 3.9-1.2A

ACTIVE VALVES (NSSS SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
RCS	FV-3651*	Makeup water supply to PRT	3	Ball	Air	2	(2)
	FV-3652*	PRT N_2 supply/vent	1	Ball	Air	2	(2)
	FV-3653*	PRT N_2 supply/vent	1	Globe	Solenoid	2	(2)
	HV-3657A,B	Reactor vessel head vent	1	Globe	Solenoid	1	(1)(4)
	HV-3658A,B	Reactor vessel head vent	1	Globe	Solenoid	1	(1)(4)
	HCV-O601	Reactor vessel head vent	1	Globe	Solenoid	2	(4)
	HCV-O602	Reactor vessel head vent	1	Globe	Solenoid	2	(4)
	PCV-655A	Pressurizer PORV	3	Globe	Solenoid	1	(1)(4)
	PCV-656A	Pressurizer PORV	3	Globe	Solenoid	1	(1)(4)
	PSV-3450	Pressurizer Safety valve	6	Safety	High Pressure	1	(1)
	PSV-3451	Pressurizer Safety valve	6	Safety	High Pressure	1	(1)
	PSV-3452	Pressurizer Safety valve	6	Safety	High Pressure	1	(1)
	XRC0001A,B	PORV isolation	3	Gate	Motor	1	(1)(4)
	XRC0046	Makeup water supply to PRT	3	Check	---	2	(2)
SI/RHR	FV-3936	RWST to SFPCCS	3	Globe	Air	2	(3)
	FV-3937	RWST to SFPCCS	3	Globe	Air	2	(3)
	FV-3970	SIS test line	3/4	Globe	Air	2	(2)
	FV-3971	SIS test line	3/4	Globe	Air	2	(2)
	FV-3983	Accumulator N_2 supply	1	Globe	Air	2	(2)
	HCV-900	Accumulator N_2 header vent	1	Globe	Solenoid	2	(4)
	HV-899	Accumulator N_2 header vent	1	Globe	Solenoid	2	(4)
	PV-3928	Accumulator N_2 supply/vent	1	Globe	Solenoid	2	(4)
	PV-3929	Accumulator N_2 supply/vent	1	Globe	Solenoid	2	(4)
	PV-3930	Accumulator N_2 supply/vent	1	Globe	Solenoid	2	(4)
	SI0011A,B,C*	HHSI pump miniflow	2	PMD	Motor	2	(3)
	SI0012A,B,C*	HHSI pump miniflow	2	PMD	Motor	2	(3)
	SI0013A,B,C*	LHSI pump miniflow	2	PMD	Motor	2	(3)
	SI0014A,B,C*	LHSI pump miniflow	2	PMD	Motor	2	(3)
	SI0058*	Accumulator N_2 supply	1	Check	---	2	(2)
	XRH0019A,B,C	LHSI pump to hot leg	8	Gate	Motor	2	(3)
	XRH0020A,B,C	LHSI pump to hot leg	8	Check	---	1	(1)(3)
	XRH0031A,B,C	LHSI pump to cold leg	8	Gate	Motor	2	(3)
	XRH0032A,B,C	LHSI pump to cold leg	8	Check	---	1	(1)(3)
	XRH0060A,B,C	RHR suction from RCS	12	Gate	Motor	1	(1)(4)
	XRH0061A,B,C	RHR suction from RCS	12	Gate	Motor	1	(1)(4)
	XRH0065A,B,C	RHR pump discharge	8	Check	---	2	(4)

TABLE 3.9-1.2A(Continued)

ACTIVE VALVES (NSSS SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
SI\RHR	XRH0066A,B	RHR letdown to CVCS	4	Gate	Motor	2	(4)
	HCV0864	RHR Hx discharge	8	Butterfly	Air	2	(3)(4)
	HCV0865	RHR Hx discharge	8	Butterfly	Air	2	(3)(4)
	HCV0866	RHR Hx discharge	8	Butterfly	Air	2	(3)(4)
	FCV0851	RHR Hx by-pass	8	Butterfly	Air	2	(3)(4)
	FCV0852	RHR Hx by-pass	8	Butterfly	Air	2	(3)(4)
	FCV0853	RHR Hx by-pass	8	Butterfly	Air	2	(3)(4)
	XSI0001A,B,C	RWST to SI pumps	16	Gate	Motor	2	(3)
	XSI0002A,B,C	RWST to SI pumps	16	Check	---	2	(3)
	XSI0004A,B,C	HHSI pump discharge	6	Gate	Motor	2	(2)(3)
	XSI0005A,B,C	HHSI pump discharge	6	Check	---	2	(2)(3)
	XSI0006A,B,C	HHSI pump to cold leg	6	Gate	Motor	2	(3)
	XSI0007A,B,C	HHSI pump to cold leg	6	Check	---	1	(1)(3)
	XSI0008A,B,C	HHSI pump to hot leg	6	Gate	Motor	2	(3)
	XSI0009A,B,C	HHSI pump to hot leg	6	Check	---	1	(1)(3)
	XSI0010A,B,C	SI pump to hot leg	8	Check	---	1	(1)(3)
	XSI0016A,B,C	Containment sump to SI pumps	16	Gate	Motor	2	(2)(3)
	XSI0018A,B,C	LHSI pump discharge	8	Gate	Motor	2	(2)(3)
	XSI0030A,B,C	LHSI pump discharge	8	Check	---	2	(2)(3)
	XSI0038A,B,C	Cold leg isolation	12	Check	---	1	(1)(3)
	XSI0039A,B,C	Accumulator discharge	12	Gate	Motor	2	(3)(4)
	XSI0046A,B,C	Accumulator discharge	12	Check	---	1	(1)(3)
CVCS	CV0022*	Letdown line containment isolation	3/4	Check	---	2	(2)
	CV0033A,B,C,D*	RCP seal water supply	2	PMD	Motor	2	(2)(4)
	CV0034A,B,C,D*	RCP seal water supply	2	Check	---	2	(2)(4)
	CV0036A,B,C,D*	RCP seal water supply	2	Check	---	1	(1)(4)
	CV0037A,B,C,D*	RCP seal water supply	2	Check	---	1	(1)(4)
	CV0077*	RCP seal water return	2	PMD	Motor	2	(2)(4)
	CV0078*	RCP seal water return	3/4	Check	---	2	(2)(4)
	CV0079*	RCP seal water return	2	PMD	Motor	2	(2)(4)
	CV0082*	Excess letdown from RCS	2	PMD	Motor	1	(1)(4)
	CV0083*	Excess letdown from RCS	2	PMD	Motor	1	(1)(4)
	LCV-0465	Normal letdown from RCS	4	Gate	Motor	1	(1)
	LCV-0468	Normal letdown from RCS	4	Gate	Motor	1	(1)
	XCV0001	Normal charging	4	Check	---	1	(1)(4)
	XCV0002	Normal charging	4	Check	---	1	(1)(4)
	XCV0003	Normal charging	4	Gate	Motor	2	(4)
	XCV0004	Alternate charging	4	Check	---	1	(1)(4)
	XCV0005	Alternate charging	4	Check	---	1	(1)(4)
	XCV0006	Alternate charging	4	Gate	Motor	2	(4)

TABLE 3.9-1.2A(Continued)

ACTIVE VALVES (NSSS SYSTEMS)

System	Valve[c] Number	Function[a]	Size (inch)	Type	Actuated By	Safety Class	ANS Active[b] Status
CVCS	XCV0023	Normal letdown containment isolation	4	Gate	Motor	2	(2)(4)
	XCV0024	Normal letdown containment isolation	4	Gate	Motor	2	(2)(4)
	XCV0025	Charging line containment isolation	4	Gate	Motor	2	(2)(4)
	XCV0026	Charging line containment isolation	4	Check	---	2	(2)(4)
	XCV0112B	VCT discharge isolation	6	Gate	Motor	2	(4)
	XCV0112C	RWST to CCPs	6	Gate	Motor	2	(4)
	XCV08377A	Centr. chg. pump #1A discharge isolation	3	Gate	Motor	2	(4)
	XCV08377B	Centr. chg. pump #1B discharge isolation	3	Gate	Motor	2	(4)
	XCV0113A	VCT discharge isolation	6	Gate	Motor	2	(4)
	XCV0113B	RWST to CCPs	6	Gate	Motor	2	(4)
	XCV0217	Boric acid tanks to CCPs	4	Check	---	2	(4)
	XCV0218	Boric acid tanks to CCPs	4	Gate	Motor	2	(4)
	XCV0224	RWST to CCPs	6	Check	---	2	(4)
	XCV0235A,B	CCP discharge	3	Check	---	2	(4)
	XCV0338	Boric acid transfer pump discharge	4	Check	---	3	(4)
	XCV0349	Boric acid transfer pump discharge	4	Check	---	3	(4)
	HCV0206	Charging line (boration) throttling	1	Globe	Solenoid	2	(4)
	XCV8348	Seal water (boration) throttling	2	Globe	Motor	2	(4)
	XCV0158	CVCS to RHR system containment isolation	4	Check	---	2	(2)
	CV0009*	Charging line aux. spray	2	Check	---	1	(1)
	XCV0639	Alternate boration path	2	Check	---	2	(4)
	FV-8400A,B	RCPC isolation	2	PMD	Air	3	(4)
	XCV0670	Charging line isolation	4	Check	---	2	(4)
	XCV0671	Seal injection isolation	2	Check	---	2	(4)
CSS							
	XCS0001A,B,C	Spray pump discharge	8	Gate	Motor	2	(2)(5)
	XCS0002	Spray pump discharge	8	Check	---	2	(2)(5)
	XCS0004	Spray pump discharge	8	Check	---	2	(2)(5)
	XCS0005	Spray pump discharge	8	Check	---	2	(2)(5)
	XCS0006	Spray pump discharge	8	Check	---	2	(2)(5)

TABLE 3.9-1.2A (Continued)

ACTIVE VALVES (NSSS SYSTEMS)

[a] Abbreviations

CCP	-	Centrifugal charging pump
CSS	-	Containment Spray System
CVCS	-	Chemical and Volume Control System
HHSI	-	High-head safety injection
LHSI	-	Low-head safety injection
PDCP	-	Positive displacement charging pump
PORV	-	Power-operated relief valve
PRT	-	Pressurizer relief tank
RCP	-	Reactor coolant pump
RCPC	-	Reactor coolant purity control
RCS	-	Reactor Coolant System
RHR	-	Residual Heat Removal
RWST	-	Refueling water storage tank
SFPCCS	-	Spent Fuel Pool Cooling and Cleanup System
SI	-	Safety injection
SIS	-	Safety Injection System
VCT	-	Volume control tank

[b] Active Status

[1] - Reactor coolant pressure boundary integrity
[2] - Containment isolation
[3] - Emergency core cooling
[4] - Cold shutdown
[5] - Containment depressurization and fission product scrubbing

[c] Valve numbers followed by an asterisk "*" are supplied by Bechtel. The remainder of the valves are supplied by Westinghouse.

TABLE 3.9-2.1

LOADING COMBINATIONS FOR ASME CLASS 1 NSSS COMPONENTS AND SUPPORTS

Condition Classification	Loading Combination
Design	Design pressure, design temperature, deadweight, Operation Basis Earthquake
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)

TABLE 3.9-2.1A

STRESS CRITERIA FOR ASME B7PV CODE, SECTION III
CLASS 1 COMPONENTS[a] (NSSS SUPPLIER)

Design/Service Level	Vessels/Tanks	Piping	Pumps	Valves	Components Supports
Design and service level A	ASME B&PV Code, Section III NB 3221, 3222	ASME B&PV Code, Section III NB 3652, 3653	ASME B&PV Code, Section III NB 3221, 3222	ASME B&PV Code, Section III NB 3520, 3525	ASME B&PV Code, Section III, Subsection NF NF 3221, 3222 NF 3231.1(a)
Service level B (UPSET)	ASME B&PV Code, Section III NB 3223	ASME B&PV Code, Section III NB 3654	ASME B&PV Code, Section III NB 3223	ASME B&PV Code, Section III NB 3525	ASME B&PV Code, Section III, Subsection NF NF 3223, 3231.1(a)
Service level C (Emergency)	ASME B&PV Code, Section III NB 3224	ASME B&PV Code, Section III NB 3655	ASME B&PV Code, Section III NB 3224	ASME B&PV Code, Section III NB 3526	ASME B&PV Code, Section III, Subsection NF NF 3224, 3231(b)
Service level D (Faulted)	ASME B&PV Code, Section III, see paragraph 3.9.1.4 NB 3225	ASME B&PV Code, Section III, see paragraph 3.9.1.4 NB 3656	ASME B&PV Code, Section III, (No active Class 1 pump used) NB 3225	See Note (b)	ASME B&PV Code, Section III, Subsection NF, see paragraph 3.9.1 NF 3225, 3221.1(c)

P_e, P_m, P_b, Q_t, C_p, S_n and S_m as defined by ASME B&PV Code, Section III

a. A test of the components may be performed in lieu of analysis.

b. CLASS 1 VALVE SERVICE LEVEL D CRITERIA

ACTIVE	INACTIVE
Calculate P_m from Subsection NB 3545.1 with internal Pressure $P_s = 1.25P_s$ $P_m \leq 1.5\ S_m$	Calculate P_m from Subsection NB 3545.1 with internal Pressure $P_s = 1.50\ P_s$ $P_m \leq 2.4S_m$ or $0.7S_u$
Calculate S_n from Subsection NB 3545.2 with $C_p = 1.5$ $P_s = 1.25P_s$ Qt2 = 0 Ped = 1.3X value of Ped from equations of 3545.2 (b)(1) $S_n \leq 3S_m$	Calculate S_n from Subsection NB 3545.2 with $C_p = 1.5$ $P_s = 1.5P_s$ Qt2 = 0 Ped = 1.3X value of Ped from equations of NB 3545.2(b)(1) $S_n \leq 3S_m$

TABLE 3.9-2.2

DESIGN LOADING COMBINATION FOR ASME CODE CLASSES 2 AND 3 COMPONENTS AND SUPPORTS (NSSS SUPPLIERS)

Condition Classification	Loading Combination
Design and Normal	Design pressure, design temperature,[1] deadweight, nozzle loads[2]
Upset	Upset condition pressure, upset condition metal temperature,[1] deadweight, OBE, nozzle loads[2]
Emergency	Emergency condition pressure, emergency condition metal temperature,[1] deadweight, nozzle loads[2]
Faulted	Faulted condition pressure, faulted condition metal temperature,[1] deadweight, SSE, nozzle loads[2]

1. Temperature is used to determine allowable stress only.
2. Nozzle loads are those loads associated with the particular plant operating conditions for the component under consideration.

TABLE 3.9-2.3

DESIGN LOADING COMBINATIONS FOR ASME III CODE
CLASS 1 COMPONENTS (BOP SCOPE OF SUPPLY)

Plant condition	Design Loading Combinations [4]
Design	PD
Normal	PD + DW + TH[1,2]
Upset	(a) PO + DW + OBE[1,3]
	(b) PO + DW + RVC + OBE[1,3]
	(c) PO + DW + FV
	(d) PO + DW + RVO + OBE[1,3]
	(e) PO + DW + DU[1]
Emergency	(a) PO + DW + DE
Faulted	(a) PO + DW + SSE
	(b) PO + DW + SSE + RVO
	(c) PO + DW + $[LOCA^2 + SSE^2]^{1/2}$
	(d) PO + DW + DF
	(e) PO + DW + HEB

1. Effects due to thermal expansion should be added for fatigue evaluation.
2. Includes loading due to thermal transient events.
3. For Class 1 fatigue evaluation, OBE includes effects due to inertially induced motions and seismic anchor motions.
4. For components other than piping, appropriate nozzle loads associated with the particular plant operating conditions are also included.

TABLE 3.9-2.3 (Continued)

DESIGN LOADING COMBINATIONS FOR ASME III CODE CLASS 1 COMPONENTS (BOP SCOPE OF SUPPLY)

DEFINITION OF TERMS

PD	- Loadings associated with the design pressure.
PO	- Loadings associated with operating pressures including, where applicable, any transient pressures associated with the loading conditions event under consideration.
DW	- Loading associated with deadweight and liveweight.
OBE	- Inertial loadings associated with the OBE.
SAM (OBE)	- Anchor point displacement loading associated with OBE earthquake.
BS	- Single nonrepeated anchor movement (building settlement).
SSE	- Inertial loading associated with the SSE.
SAM (SSE)	- Anchor point displacement loading associated with SSE.
RVC	- Transient loadings associated with relief valve blowdown in a closed system.
RVO	- Sustained loadings associated with relief valve in an open system.
FV	- Transient loadings associated with fast valve closure.
TH	- Loadings associated with thermal expansion.
LOCA	- Loss of coolant accident - defined as those postulated accidents that result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant make-up system, from breaks in the reactor coolant pressure boundary up to and including a break equivalent in size to the rupture of the largest branch pipe connected to the reactor coolant LOOP. This condition includes the loads from the postulated pipe break itself and also any associated system transients or dynamic effects resulting from the postulated pipe break.

TABLE 3.9-2.3 (Continued)

DESIGN LOADING COMBINATIONS FOR ASME III CODE CLASS 1 COMPONENTS (BOP SCOPE OF SUPPLY)

DEFINITION OF TERMS

HEB	- Loadings associated with high-energy line pipe breaks (includes loadings from jet impingement, pipe motion, and pipe impact).
DU	- Loadings associated with other transient dynamic event classified as an upset condition.
DE	- Loadings associated with other transient dynamic event classified as an emergency condition.
DF	- Loadings associated with other transient dynamic events classified as faulted conditions.

TABLE 3.9-2.3A

DESIGN LOADING COMBINATIONS FOR ASME III CODE CLASS 2 AND 3 COMPONENTS (BOP SCOPE OF SUPPLY)

Plant Condition	Design Loading Combinations [1, 2, 3]	
Design	PD	
Normal	PD + DW	
Upset	(a)	PO + DW + OBE[3]
	(b)	PO + DW + RVC
	(c)	PO + DW + FV
	(d)	PO + DW + OBE[3] + RVO
	(e)	PO + DW + DU
Emergency	(a)	PO + DW + DE
Faulted	(a)	PO + DW + SSE
	(b)	PO + DW + SSE + RVO
	(c)	PO + DW + HEB
	(d)	PO + DW + DF
	(e)	P0 + DW + LOCA
Thermal	(a)	TH + SAM (OBE)[3]

1. Other loads such as loads due to building settlement, displacement of containment due to design basis accident (DBA) and wind loads are also considered where applicable.
2. For definition of terms see Table 3.9-2.3.
3. Loading effects due to seismic anchor motions are either included with OBE or added to thermal effects (TH).
4. For components other than piping, appropriate nozzle loads associated with the particular plant operating conditions are also included.

TABLE 3.9-2.4

DESIGN LOADING COMBINATIONS FOR ASME III CODE
CLASS 1, 2 AND 3 COMPONENT SUPPORT (BOP SCOPE OF SUPPLY)

Plant Condition	Design Loading Combinations [1, 2, 3]
Design	DW
Normal	DW + TH
Upset	(a)DW + TH + $[OBE^2 + SAM^2 (OBE)]^{½}$ (b)DW + TH + $[OBE^2 + SAM^2 (OBE)]^{½}$ + RVO (c)DW + TH + RVC (d)DW + TH + FV (e)DW + TH + RVO (f)DW + TH + DU
Emergency	(a)DW + TH + DE
Faulted	(a)DW + TH + $[SSE^2 + SAM^2 (SSE)]^{½}$ (b)DW + TH + $[SSE^2 + SAM^2 (SSE)]^{½}$ + RVO (c)DW + TH + DF (d)DW + TH + $[SSE^2 + SAM^2 (SSE) + LOCA^2]^{½}$ [4] (e)DW + TH + HEB (f)DW + TH + LOCA

1. For definition of terms see Table 3.9-2.3.
2. Other loads such as loads due to building settlement, displacement of containment due to design basis accident (DBA) and wind loads are also considered where applicable.
3. Pressure loads due to unbalanced expansion joints are considered for normal, upset, emergency, and faulted conditions where applicable.
4. This loading combination applies to Class 1 component supports and supports on Class 2 piping analyzed with Class 1 piping.

TABLE 3.9-2.4A

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER SAFETY AND RELIEF VALVE PIPING AND SUPPORTS (ASME CLASS 1 PORTION)

Condition Classifications	Load Combination[(2)]	Service Limit
Design	Pressure, Weight	Design
Normal	Condition Transients, Weight	Level A
Upset	Conditions Transients, Weight OBE[(2)], VT[(3)]	Level B
Emergency	Conditions Transients, Weight, VT[(3)]	Level C
Faulted	Condition Transients, Weight, SSS[(1)], VT[(3)]	Level D

1. The OBE and SSE loadings include the effects of seismic anchor motions.
2. Dynamic loads are combined by SRSS.
3. Valve thrust loads (VT) are loads resulting from the rapid acceleration or deceleration of a water mass, non-condensable gases, or both.

TABLE 3.9-2.4A (Continued)

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER
SAFETY AND RELIEF VALVE PIPING AND SUPPORTS
(B31.1 SEISMICALLY DESIGNED DOWNSTREAM PORTION)

Condition Combination[(2)]	Load Combination
Design	Design Pressure, Weight
Normal	Normal Condition Transients, Weight
Upset	Upset Condition Transients, Weight OBE [(1)], VT [(3)]
Emergency	Emergency Condition Transients, Weight VT[(3)]
Faulted	Faulted Condition Transients, Weight SSE[(1)], VT[(3)]

1. The OBE and SSE loadings include the effects of seismic anchor motions.
2. Dynamic loads are combined by SRSS.
3. Valve thrust loads (VT) are loads resulting from the rapid acceleration or deceleration of a water mass, non-condensable gases, or both.

TABLE 3.9-2.5

CONFORMANCE WITH NRC REGULATORY GUIDE 1.48
(BOP SCOPE OF SUPPLY)

RG 1.48 Position		STPEGS Position
C.1	ASME Code Class-1	In compliance Vessels and Piping
C.2	Non-active code Class-1 Pump and Valves designed by analysis.	In compliance for Class-1 valves. Class-1 pumps are not in BOP scope of supply.
C.3	Non-active ASME Code Class-1 valves designed by standard or alternate rules.	In compliance
C.4	Active ASME Code Class-1 Pumps and Valves designed by analysis.	In compliance
C.5	Active ASME Code Class-1 valves designed by standard or alternate design rules.	In compliance
C.6	ASME Code Class-2 and 3 vessels designed to Div. 1 of Section VIII of ASME Code.	STPEGS does not have ASME Code Class 2 and 3 vessels which are designed to ASME Section VIII, Div. 1.
C.7	ASME Code Class-2 vessels designed to Div. 2 of Section VIII of the ASME Code.	STPEGS does not have ASME Class-2 vessels designed to ASME Section VIII, Div. 2.
C.8	ASME Code Class 2 and 3 Piping.	STPEGS uses stress limits specified in ASME Code Case 1606-1 (N-53). This Code Case was adopted in Winter 1976 Addenda of the ASME III Code, Subsections NC-3611.2 and ND-3611.2.
C.9	Non-active ASME Code Class 2 and 3 Pumps	STPEGS uses higher stress limits specified in ASME Code Case 1636-1 (N-70). This Code Case was adopted in Winter 1976 Addenda of the ASME III Code, Subsection-NC-3400 and ND-3400.

TABLE 3.9-2.5 (Continued)

CONFORMANCE WITH NRC REGULATORY GUIDE 1.48
(BOP SCOPE OF SUPPLY)

RG 1.48 Position	STPEGS Position
C.10 Active ASME Code Class 2 and 3 Pumps	STPEGS uses higher stress limits allowed by note 11 of Reg. Guide 1.48. In accordance with this note STPEGS uses the stress limits for non-active pumps specified in Reg. Guide 1.48 position C.9 and assures by analysis and/or testing that the operability of the pump is not impaired when designed to these limits.
C.11 Non-active Code Class 2 and 3 Valves	STPEGS uses higher stress limits specified in ASME Code Case 1635-1 (N-69). This Code Case was adopted in Winter 1976 Addenda of ASME Code, Subsection NC-3500 and ND-3500.
C.12 Active ASME Code Class 2 and 3 Valves	STPEGS uses higher stress limits specified in ASME Code Case 1635-1 (N-69) which are the same limits used for non-active valves. Note 11 of Reg. Guide 1.48 allows the use of stress limits for non-active valves provided appropriate analysis and/or testing confirms that operability of the valve is not impaired when designed to these limits. STPEGS provides this operability assurance.

TABLE 3.9-3

STRESS CRITERIA FOR ASME III CODE
CLASS 2 AND 3 VESSELS DESIGNED TO NC-3300 AND ND-3300
(BOP SCOPE OF SUPPLY)

Plant Condition	Stress Limits[1,2]
Design and Normal	$\sigma m \leq 1.0S$ $(\sigma m \text{ or } \sigma L) + \sigma b \leq 1.5S$
Upset	$\sigma m \leq 1.1S$ $(\sigma m \text{ or } \sigma L) + \sigma b \leq 1.65S$
Emergency	$\sigma m \leq 1.5S$ $(\sigma m \text{ or } \sigma L) + \sigma b \leq 1.8S$
Faulted	$\sigma m \leq 2.0S$ $(\sigma m \text{ or } \sigma L) + \sigma b \leq 2.4S$

Definition of Terms:

σm	=	General membrane stress
σL	=	Local membrane stress
σb	=	Bending stress
S	=	Allowable stress value in accordance with ASME III Code
OBE	=	Operating Basis Earthquake
SSE	=	Safe Shutdown Earthquake

1. These stress limits do not take into account either local or general buckling which might occur in the thin wall vessels.
2. Nozzle loads are provided in the equipment specification and are considered in the design. Piping system design limits the loads imposed on the nozzles to those values.

TABLE 3.9-3A

STRESS CRITERIA FOR ASME CODE CLASS 2 AND 3
TANKS BY NSSS SUPPLIER[1]

Design/Service Level	Stress Limits[2]
Design and Service Level A	NC-3217
Service Level B	NC-3217
Service Level C	NC-3217
Service Level D	NC-3217

1. Applies for tanks designated in accordance with ASME B&PV Code, Section III, NC-3200.
2. As specified by ASME B&PV Code, Section III.

TABLE 3.9-4

STRESS CRITERIA FOR ASME CODE SECTION III
CLASS 2 AND 3 NONACTIVE PUMPS[5]
(BOP SCOPE OF SUPPLY)

Plant Condition	Stress Limits[2,3,4]	Pmax[1]
Design and Normal	The pump conforms to the requirements of ASME Code, Section III, NC-3400 (or ND-3400)	1.0
Upset	$\sigma m \leq 1.1S$ $(\sigma m \text{ or } \sigma L) + \sigma b \leq 1.65S$	1.1
Emergency	$\sigma m \leq 1.5S$ $(\sigma m \text{ or } \sigma L) + \sigma b \leq 1.8S$	1.2
Faulted	$\sigma m \leq 2.0S$ $(\sigma m \text{ or } \sigma L) + \sigma b \leq 2.4S$	1.5

1. The maximum pressure does not exceed the tabulated factors listed under Pmax times the design pressure.
2. Pump mounting feet or pedestal supports which are integrally attached to the pump pressure retaining boundary are designed to the stress limit requirements in this table. Pump support components which are not integrally attached to the pump pressure retaining boundary are designed to the requirements of ASME III, subsection NF and AISC as applicable.
3. Loads imposed on the pump nozzles by connecting piping are provided in the pump specification and are considered in the pump design. Piping system design limits the loads imposed on the pump nozzles to those values.
4. Definition of terms is provided in Table 3.9-3.
5. DG lube oil circulation pump and DG jacket water circulation pump.

TABLE 3.9-4A

STRESS CRITERIA FOR ASME CODE SECTION III
CLASS 2 AND 3 ACTIVE PUMPS
(BOP SCOPE OF SUPPLY)

Plant Condition	Stress Limits[2,3,4]	Pmax[1]
Design and Normal	The pump conforms to the requirements of ASME Code, Section III, NC-3400 (or ND-3400)	1.0
Upset	$\sigma m \leq 1.0S$ $(\sigma m \text{ or } \sigma L) + \sigma b \leq 1.50S$	1.1
Emergency	$\sigma m \leq 1.2S$ $(\sigma m \text{ or } \sigma L) + \sigma b \leq 1.65S$	1.2
Faulted	$\sigma m \leq 1.2S$ $(\sigma m \text{ or } \sigma L) + \sigma b \leq 1.8S$	1.5

1. The maximum pressure does not exceed the tabulated factors listed under Pmax times the design pressure.
2. For active pumps, in addition to compliance with the design limits specified in this table, assurance of pump operability under all design loading combinations is provided by testing and/or detailed stress analysis as required in the pump specification.
3. Pump mounting feet or pedestal supports which are integrally attached to the pump pressure retaining boundary are designed to the stress limit requirements in this table. Pump support components which are not integrally attached to the pump pressure retaining boundary are designed to the requirements of ASME III, Subsection NF and AISC as applicable.
4. Loads imposed on the pump nozzles by connecting piping are provided in the pump specification and are considered in the design. Piping system design limits the loads imposed on the pump nozzles to those values.
5. Definition of terms is provided in Table 3.9-3.

TABLE 3.9-4B

STRESS CRITERIA FOR ASME CODE CLASS 2 AND 3 ACTIVE PUMPS BY THE NSSS SUPPLIER

Design/Service Level	Stress Limits[1]
1.Design and Service Level A	ASME B&PV Code, Section III, Sub-section NC-3400 and ND-3400
2.Service Level B	$\sigma m \leq 1.0$ S $\sigma m + \sigma b \leq 1.5$ S
3.Service Level C	$\sigma m \leq 1.2$ S $\sigma m + \sigma b \leq 1.65$ S
4.Service Level D	$\sigma m \leq 1.2$ S $\sigma m + \sigma b \leq 1.8$ S

1. For nomenclature refer to ASME B&PV Code, Section III, NC-3423 (ND-3423).

TABLE 3.9-4C

STRESS CRITERIA FOR ASME CODE CLASS 2 AND 3
NON-ACTIVE PUMPS[1] BY THE NSSS SUPPLIER

Design/Service Level	Stress Limits[2]
Design and Service Level A	NC-3423 (ND-3423)
Service Level B	NC-3423 (ND-3423)
Service Level C	NC-3423 (ND-3423)
Service Level D	NC-3423 (ND-3423)

1. Positive Displacement Charging Pump,
   Reactor Coolant Purification Pump,
   Recycle Evaporator Feed Pump,
   Refueling Water Purification Pump.
2. As specified by ASME B&PV Code, Section III.

TABLE 3.9-5

STRESS CRITERIA FOR ASME III CODE
CLASS 1 VALVES (BOP SCOPE OF SUPPLY)

Plant Condition	Stress Limits[1]
Design and Normal	ASME III, NB-3500 (Standard Design Rules)
Upset	ASME III, NB-3525
Emergency	ASME III, NB-3526
Faulted	ASME III, NB-3527

1. For active valves, in addition to compliance with the design limits specified in this table, assurance of valve operability under all design loading combinations is provided by testing and/or detailed stress and deformation analyses as specified in the valve specifications.

TABLE 3.9-5A

STRESS CRITERIA FOR ASME CODE SECTION III
CLASS 1 VALVES (NSSS)

Condition	Stress Limits [1-4]	Pmax [5]
Design and Normal	Conform to the requirements of ASME Code, Sec. III, NB-3500	1.0
Upset	$Pm \leq Sm$ $Pm + Pb \leq 1.5\ Sm$	1.1
Emergency	Conform to the requirements of ASME Code Sec. III, NB-3500	1.2
Faulted	$Pm \leq 2.4\ Sm$ $Pm + Pb \leq 1.5\ Sm$ $Sn \leq 3.0\ Sm$	1.5

Pm, Pb, Sn, Sm, are as defined by ASME Section III, Subsection NB.

TABLE 3.9-5A (Continued)

STRESS CRITERIA FOR ASME III CODE
CLASS 2 AND 3 VALVES (BOP SCOPE OF SUPPLY)

Notes:

1. Valve nozzle (piping load) stress analysis is not required when both of the following conditions are satisfied by calculation: (a) section modulus and area of plane, normal to the flow through the region of valve body crotch is at least 10 percent greater than the piping connected (or joined) to the valve body inlet and outlet nozzles; and (b) code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, of connected piping material. If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated in (a) above shall be multiplied by the ratio of the allowable stress for the pipe divided by the allowable stress of the valve. If unable to comply with these requirements, the design by analysis procedure of NB-3545.2 is an acceptable alternate method.

2. These stress limits are applicable to the pressure-retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

3. Design requirements listed in this table are not applicable to valve discs, stems, set 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.

4. These rules do not apply to Class 2 and 3 safety and relief valves.

5. The maximum pressure does not exceed the tabulated factors listed under Pmax times the design pressure or times the rated pressure at the applicable service temperature.

6. For Active Valves, in addition to compliance with the above design limits, assurance of valve operability under all design loading combinations is provided by testing and/or analysis as required in the valve specification.

TABLE 3.9-6

STRESS CRITERIA FOR SAFETY RELATED ASME CODE
CLASS 2 AND 3 VALVES BY NSSS SUPPLIER

Design/Service Level	Stress Limits[1]
Design and Service Level A	NC-3510 (ND-3510)
Service Level B	NC-3520 (ND-3520)
Service Level C	NC-3520 (ND-3520)
Service Level D	NC-3520 (ND-3520)

1. As specified by ASME B&PV Code, Section III.

TABLE 3.9-6A

STRESS CRITERIA FOR ASME III CODE
CLASS 2 AND 3 VALVES (BOP SCOPE OF SUPPLY)

Plant Condition	Stress Limits[1,2,3,4,5]	Pmax
Design and Normal	ASME III, NC-3500 or ND-3500	1.0
Upset	$\sigma m \leq 1.1S$ $(\sigma m \text{ or } \sigma L) + \sigma b \leq 1.65S$	1.1
Emergency	$\sigma m \leq 1.5S$ $(\sigma m \text{ or } \sigma L) + \sigma b \leq 1.8S$	1.2
Faulted	$\sigma m \leq 2.0S$ $(\sigma m \text{ or } \sigma L) + \sigma b \leq 2.4S$	1.5

Definition of Terms:

σm	=	General membrane stress
σL	=	Local membrane stress
σb	=	Bending stress
S	=	Allowable stress value in accordance with ASME III Code
Pmax	=	Maximum internal pressure (See Note 5)
OBE	=	Operating Basis Earthquake
SSE	=	Safe Shutdown Earthquake

TABLE 3.9-7

STRESS CRITERIA FOR ASME III CODE
CLASS 1 PIPING
(BOP SCOPE OF SUPPLY)

Plant Condition	Stress Limits[1]
Design	NB-3221 or NB-3652
Normal	NB-3222 or NB-3653
Upset	NB-3223 or NB-3654
Emergency	NB-3224 or NB-3655
Faulted	NB-3225 or NB-3656
Testing	NB-3226 or NB-3657

1. As specified by ASME Section III, 1974 and Addenda through Winter 1975 and NB-3650 and NB-3680 of Summer 1979 Addenda.

TABLE 3.9-7A

STRESS CRITERIA FOR ASME III CODE
CLASS 2 AND 3 PIPING
(BOP SCOPE OF SUPPLY)

Plant Condition	Stress Limits[1]
Design and Normal	ASME III, NC-3600 and ND-3600
Upset, Emergency and Faulted	The piping shall conform to the requirements of ASME Code Case 1606-1 (N-53)[2]

1. As specified by ASME Section III, 1974 and Addenda through Winter 1975.
2. Code Case 1606-1 (N-53) was adopted in Winter 1976 addenda of the ASME III Code, Subsection NC-3611.2 and ND-3611.2.

TABLE 3.9-7B

STRESS CRITERIA FOR ASME III CODE
CLASS 1 COMPONENT SUPPORTS
(BOP SCOPE OF SUPPLY)

Support Type	Plant Condition and Stress Limits[1]				
	Design	Normal	Upset	Emergency	Faulted
Plate and shell design by analysis	NF-3221	NF-3222	NF-3223	NF-3224	NF-3225
Linear type supports design by analysis	NF-3231	NF-3231	NF-3231	NF-3231	NF-3231
Component standard supports, design by analysis	NF-3240	NF-3240	NF-3240	NF-3240	NF-3240
Component supports, design by load rating	NF-3260	NF-3260	NF-3260	NF-3260	NF-3260

1. Paragraph numbers refer to ASME Code, Section III, Subsection NF.

TABLE 3.9-7C

STRESS CRITERIA FOR ASME III CODE
CLASS 2 AND 3 COMPONENT SUPPORTS
(BOP SCOPE OF SUPPLY)

Support Type	Plant Condition and Stress Limits[1]				
	Design	Normal	Upset	Emergency	Faulted
Plate and shell, design by analysis	NF-3321	NF-3321	NF-3321	NF-3321	NF-3321
Linear type supports design by analysis	NF-3330	NF-3330	NF-3330	NF-3330	NF-3330
Component standard supports, design by analysis	NF-3340	NF-3340	NF-3340	NF-3340	NF-3340
Component supports, design by load rating	NF-3360	NF-3360	NF-3360	NF-3360	NF-3360

1. Paragraph numbers refer to ASME Code, Section III, Subsection NF.

TABLE 3.9-8

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 percent of full power per minute	Unit 1 – 3000 (each) (2) Unit 2 – 10,300 (each) (1)
3.	Step load increase and decrease of 10 percent of full power	2,000 (each)
4.	Large stop load decrease with steam dump	200
5.	Steady state fluctuations	
	a. Initial fluctuations	1.5×10^5
	b. Random fluctuations	3.0×10^6
6.	Feedwater cycling at hot shutdown	2,000
7.	Loop out of service	
	a. Normal loop shutdown	80
	b. Normal loop startup	70
8.	Unit loading and unloading between 0 to 15 percent of full power	500 (each)
9.	Boron concentration equalization	26,400
10.	Refueling	80
11.	Primary side leak test	Unit 1 – 120 (2) Unit 2 – 200
12.	Secondary side leak test	80
13.	Tube leak test	800
14.	Turbine roll test	20

TABLE 3.9-8 (Continued)

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

Upset Conditions	Occurrences
1. Loss of load, without immediate	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
a. Inadvertent auxiliary spray	10
6. Inadvertent startup of an inactive loop	10
7. Control rod drop	80
8. Inadvertent ECCS actuation	60
9. Operating Basis Earthquake (5 earthquakes of 10 cycles each)	50 cycles
10. Excessive feedwater flow	30
11. RCS Cold Overpressurization	10

TABLE 3.9-8 (Continued)

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

Emergency Conditions*	Occurrences
1. Small LOCA	5
2. Small steam break	5
3. Complete loss of flow	5

Faulted Conditions*	Occurrences
1. Reactor coolant branch pipe break (LOCA)	1
2. Large steam 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 Condition 4.c)
7. Safe Shutdown Earthquake	1

Test Conditions	Occurrences
1. Primary side hydrostatic test	Unit 1 - 1(2) Unit 2 - 10
2. Secondary side hydrostatic test	10

* In accordance with ASME B&PV Code Section III, emergency and faulted conditions are not included in the fatigue evaluation.

(1) Limit on main feedwater control valve only.
(2) Limited by BMI Half Nozzle Repair. (Reference: DCP 03-6248-16 Supp. 9)

TABLE 3.9-9

MAXIMUM DEFLECTIONS ALLOWED FOR REACTOR INTERNAL SUPPORT STRUCTURES

Component	Allowable Deflections (in.)	No-Loss-of-Function Deflections
Upper Barrel		
radial inward	4.1	8.2
radial outward	1.0	1.0
Upper Package	0.10	0.15
Rod Cluster Guide Tubes	1.0	1.4

TABLE 3.9-10

SAFETY-RELATED SEISMIC
CATEGORY I EQUIPMENT QUALIFICATION PROCEDURE
(NSSS SCOPE)

Component	Safety Class	Seismic Qualification Procedure	Remarks
Reactor Coolant System			
Reactor Vessel	1	Analysis	
CRDM Assemblies	1	Analysis	
CRDM Head Adapter Plugs	1	Analysis	
Steam Generators			
(tube side)	1	Analysis	
(shell side)	2	Analysis	
Pressurizer	1	Analysis	
Reactor Coolant Hot and Cold Leg Piping, Supports, Fittings	1	Analysis	
Surge Pipe, Supports, Fittings	1	Analysis	
Crossover Leg Piping, Fittings	1	Analysis	
Pressurizer Safety Valves	1	Analysis & Test	
Pressurizer Power-Operated Relief Valves	1	Analysis & Test	
Valves of Safety Class 1 to Safety Class 2 Interface	1	Analysis & Test	
Reactor Coolant Pump			
RCP Casing	1	Analysis	
Main Flange	1	Analysis	
Thermal Barrier	1	Analysis	
Thermal Barrier HX	1	Analysis	
No. 1 Seal Housing	1	Analysis	

TABLE 3.9-10 (Continued)

SAFETY-RELATED SEISMIC
CATEGORY I EQUIPMENT QUALIFICATION PROCEDURE
(NSSS SCOPE)

Component	Safety Class	Seismic Qualification Procedure	Remarks
Reactor Coolant System (Continued)			
No. 2 Seal Housing	2	Analysis	
Pressure-Retaining Bolting	1	Analysis	
RCP Motor Rotor	2	Analysis	
RCP Motor Shaft and Coupling	2	Analysis	
Spool Piece	2	Analysis	
Armature	2	Analysis	
Flywheel	2	Analysis	
Motor Bolting	2	Analysis	Applies only to bolting involved with coastdown function
Motor Stand	2	Analysis	
Motor Frame	2	Analysis	
Bearings (upper thrust)	2	Analysis	
Upper Oil Cooler Reservoir			
(tube side)	3	Analysis	
(shell side)	3	Analysis	
Lower Oil Cooler Reservoir			
(tube side)	3	Analysis	
(shell side)	3	Analysis	
Oil Cooler Piping	3	Analysis	

TABLE 3.9-10 (Continued)

SAFETY-RELATED SEISMIC
CATEGORY I EQUIPMENT QUALIFICATION PROCEDURE
(NSSS SCOPE)

Component	Safety Class	Seismic Qualification Procedure	Remarks
Reactor Coolant System (Continued)			
Motor Air Coolers	3	Analysis	Applies only to portions containing component cooling water
Thermocouple Column Seal Assembly	1	Analysis	
Thimble Guide Tubing	1	Analysis*	
Seal Table & Parts	1	Analysis	
Flux Thimble Assembly	1	Analysis	
RCS Components Supports	1	Analysis	
Hydraulic Shock Suppressor	1	Analysis	
Pressurizer Safety & Relief Valve Piping and Supports	1	Analysis	
Missile Shield	1	Analysis	
Reactor Vessel Head Lift Rods (portions that support (CRDMs)	1	Analysis	
Reactor Vessel Shoes and Shims	1	Analysis	
Chemical and Volume Control System			
Regenerative HX	2	Analysis	
Letdown HX			
(tube side)	2	Analysis	
(shell side)	3	Analysis	

TABLE 3.9-10 (Continued)

SAFETY-RELATED SEISMIC
CATEGORY I EQUIPMENT QUALIFICATION PROCEDURE
(NSSS SCOPE)

Component	Safety Class	Seismic Qualification Procedure	Remarks
Chemical and Volume Control System (Continued)			
Reactor Coolant Filter	2	Analysis	
Volume Control Tank	2	Analysis	
Centrifugal Charging Pump	2	Analysis	
Centrifugal Charging Pump Motor	2	Test	
Positive Displacement Pump	2	Analysis	
Positive Displacement Pump Motor	2	Test	
Reactor Coolant Purification Pump	3	Analysis	
Reactor Coolant Purification Pump Motor	3	Test	
Seal Water Injection Filter	2	Analysis	
Letdown Orifices	2	Analysis	
Excess Letdown HX			
(tube side)	2	Analysis	
(shell side)	2	Analysis	
Seal Water Return Filter	2	Analysis	
Seal Water HX (tube side)	2	Analysis	
(shell side)	3	Analysis	
Boric Acid Transfer Pump	3	Analysis	
Boric Acid Transfer Pump Motor	3	Test	
Boric Acid Filter	3	Analysis	

TABLE 3.9-10 (Continued)

SAFETY-RELATED SEISMIC
CATEGORY I EQUIPMENT QUALIFICATION PROCEDURE
(NSSS SCOPE)

Component	Safety Class	Seismic Qualification Procedure	Remarks
Chemical and Volume Control System (Continued)			
Reactor Coolant Pump Seal Bypass Orifices	1	Analysis	
Mixed Bed Demineralizer	3	Analysis	
Cation Bed Deminalizer	3	Analysis	
Letdown Filter	3	Analysis	
Boron Thermal Regeneration Subsystem			
Letdown Reheat HX (tube side)	2	Analysis	
Emergency Core Cooling System			
Accumulators	2	Analysis	
HHSI Pumps	2	Analysis	
HHSI Pump Motor	2	Test	
LHSI Pumps	2	Analysis	
LHSI Pump Motor	2	Test	
Residual Heat Removal System			
Residual Heat Removal Pump	2	Analysis	
Residual Heat Removal Pump Motor	2	Test	
Residual HX			
(tube side)	2	Analysis	
(shell side)	3	Analysis	

TABLE 3.9-10 (Continued)

SAFETY-RELATED SEISMIC
CATEGORY I EQUIPMENT QUALIFICATION PROCEDURE
(NSSS SCOPE)

Component	Safety Class	Seismic Qualification Procedure	Remarks
Spent Fuel Pool Cooling and Cleanup System			
Spent Fuel Pool Heat Exchanger (tube side)	3	Analysis	
(shell side)	3	Analysis	
Spent Fuel Pool Pumps	3	Analysis	
Spent Fuel Pool Pump Motor	3	Test	
Main Steam System			
Main Steam Isolation Valves	2	Analysis & Test	
Steam Power Operated Relief Valves	2	Analysis & Test	
Fuel Handling System			
Fuel Handling Machine	3	Analysis	
Telescoping Fuel-Handling Tool	3	Analysis & Test	
Spent Fuel Handling Tool	3	Analysis & Test	
Fuel Transfer System			
Fuel Transfer Tube and Flange	2	Analysis	
New Fuel Racks	3	Analysis	
Spent Fuel Racks	3	Analysis	
Neutron Detector Positioners	2	Test	

TABLE 3.9-10 (Continued)

SAFETY-RELATED SEISMIC
CATEGORY I EQUIPMENT QUALIFICATION PROCEDURE
(NSSS SCOPE)

Component	Safety Class	Seismic Qualification Procedure	Remarks
Containment Spray System			
Spray Additive Tanks	3	Analysis	
Containment Spray Pump	2	Analysis	
Containment Spray Pump Motor	2	Test	
Containment Spray Eductor	2	Analysis	
Containment Spray Nozzles	2	Analysis	
Reactor Vessel or Core Related			
Irradiation Sample Holder	2	Analysis	
CRDM Seismic Support Tie Rod Assemblies	1	Analysis	
Reactor Vessel Internals	2	Analysis & Test	
Control Rod Clusters	2	Analysis	

* Analysis performed by balance of plant.

TABLE 3.9-18

REACTOR VESSEL SUPPORT LOADS AND STRESSES

Loading Condition	Vertical Load (kips)	Horizontal Load (kips)	Max. Stress - Percent of Allowable
Normal	834	- - -	< 97
Upset	1067	542	97
Faulted	2521	1610	80

TABLE 3.9-21

COMPARISON OF RESULTS BETWEEN ME 913 AND ASME SAMPLE PROBLEM[c]

		Equations						
Components	Programs	Eq (9) Stress[a]	Eq (10) Stress	Eq (11) Stress	Eq (12) Stress	Eq (13) Stress	Eq (14) Stress	Total Usage Factor
Girth butt weld	ME 913	25,950	63,112	111,833	6,563	49,526	55,917	0.0555
(Location 19)	Sample Problem	23,400	52,549	80,677	[b]	[b]	40,338	0.0126
Butt weld tee	ME 913	24,600	65,567	128,920	39,536	23,152	135,937	0.3439
(Location 10)	Sample Problem	23,400	65,596	128,950	39,564	23,155	135,977	0.3699

a. All stresses are in psi.
b. Because S_n, calculated by equation (10), is less than $3S_m$, (52,800 psi for type 304 at 400°F), equations (12) and (13) are satisfied.
c. Sample analysis of a Class 1 piping system, prepared by the working group on piping (SGD, SC3) of the ASME BP&V Code, December 1971.

TABLE 3.9-22

HAND-CALCULATED RESULTS COMPARED WITH ME 913 OUTPUTS

Components	Programs	Equations						
		Eq (9) Stress[a]	Eq (10) Stress	Eq (11) Stress	Eq (12) Stress	Eq (13) Stress	Eq (14) Stress	Total Usage Factor
Tee	ME 913 outputs	23,650	87,446	154,318	27,805	20,433	164,256	0.7745
(Location 115)	Hand Calculated Results	23,646	87,358	154,270	27,789	20,419	164,220	[b]

a. All stresses are in psi.
b. The total usage factor in hand calculated column is left empty because of large involvements in hand computations.

TABLE 3.9-23

COMPLIANCE WITH REGULATORY GUIDE 1.148 (NSSS SCOPE)

The guidance contained in Regulatory Guide 1.148 is followed for NSSS-supplied valves with the following clarifications:

1. In general, the requirements delineated by ANSI N278.1-1975 are addressed as part of the design specification for power-actuated valves designed in accordance with ASME Section III, Class 1, 2, and 3.

2. Paragraph C.1.a. - No manually operated valves are classified as active. Active valves consist of selected relief valves, check valves, and power-actuated valves.

3. Paragraph C.1.b. - The NSSS active valve specifications are performance specifications, which are considered a "complete basis for construction." ASME Code Class 1, 2, and 3 valve specifications include design information as required by NCA-3250 of the ASME Code. However, these specifications address installation by reference to drawings and do not include maintenance requirements. Maintenance requirements are included in technical manuals.

4. Paragraph C.1.c. - For active valves, project design specifications are prepared per valve type, which meet the applicable ASME Code requirements and contain the functional requirements necessary to ensure that the valve will perform its intended safety function.

5. Paragraph C.2.a.(2) - Valve specifications do not identify the relationship between the "application characteristics" of the Code.

6. Paragraph C.2.b.(1) and (2) - In general, valve specifications do not completely address the interdependence and number of cycles of the parameters cited or the time relationship between seismic loadings and other loadings.

7. Paragraph C.2.b.(3) - These requirements are provided for active valves in the form of accelerations and nozzle loads which must be met in piping analyses.

8. Paragraph C.2.b.(4) - The maximum differential pressure is provided in specifications for power-actuated valves. Check valve specifications identify the operating and design conditions, but not the maximum differential pressure. The design condition for check valves implies a maximum differential pressure that corresponds to the ANSI pressure-temperature rating in the specifications. Water hammer and other plant conditions are analyzed for selective active valves, if required, but this information is not part of the specification.

TABLE 3.9-23 (Continued)

COMPLIANCE WITH REGULATORY GUIDE 1.148 (NSSS SCOPE)

9. Paragraph C.2.c.(2) - Valve specifications generally do not state whether the valve safety function applies to events defined in the plant operational modes or in the transient and accident classification. The appropriate service conditions are provided in the specification along with the actual valve assembly operation (open, close, or regulate fluid flow).

TABLE 3.9-24

COMPLIANCE WITH REGULATORY GUIDE 1.148 (NON-NSSS SCOPE)

The guidance contained in Regulatory Guide 1.148 is followed for non-NSSS supplied valves with the following clarifications:

1. In general, guidance contained in ANSI N278.1-1975 is addressed as part of the design specification for safety-related ASME Section III active valves.

2. Paragraph C.1.a. - No manually operated valves are classified as active.

3. Paragraph C.1.b. - The BOP active valve specifications are performance specifications which are considered a "complete basis for construction." The valve specifications contain design information as required by NCA-3250 of ASME Code. Installation and maintenance requirements are included in technical manuals and drawings.

4. Paragraph C.1.d. - There are no quality group D active valves.

5. Paragraph C.2.a(2). - Valve specifications do not identify the relationship between the "application characteristics" of the Code.

6. Paragraph C.2.b(1). - Valve specifications do not address the interdependence and number of cycles of the parameters cited.

7. Paragraph C.2.b(2). - The time relationship between seismic loadings and other applied loadings is not addressed in the valve specifications. However, the loading combinations conservatively add these loads as concurrent conditions.

8. Paragraph C.2.b(3). - These requirements are provided for active valves in the form of accelerations and nozzle loads which must be met in the piping design and analyses.

9. Paragraph C.2.b(4). - The maximum differential pressure is provided for all active valves, except check valves. Check valve specifications identify the operating and design conditions, but not the maximum differential pressure. The design specification for check valves requires hydrostatic testing for the maximum allowable pressure for that valve class. Water hammer and other plant conditions are analyzed as appropriate. This information is checked against the design specification, but is not specifically included.

10. Paragraph C.2.c.(1). - Motor operated valves are understood to fail as is. It is not specified.

TABLE 3.9-24 (Continued)

COMPLIANCE WITH REGULATORY GUIDE 1.148 (NON-NSSS SCOPE)

11. Paragraph C.2.c.(2). - Valve specifications generally do not state whether the valve safety function applies to events defined in the plant operational modes or in the transient and accident classification. The appropriate service conditions are provided in the specification along with the actual valve assembly operation (open, close, or regulate fluid flow).

12. Paragraph C.2.d. - Seat leakage limits are established for valves based on the specific safety functions and are identified in the specification.

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I MECHANICAL AND ELECTRICAL EQUIPMENT

Seismic Category I mechanical and electrical equipment, instrumentation and supports are identified in the Seismic Master List Submittal. The information to demonstrate that they are capable of performing their designated safety-related functions in the event of an earthquake is presented in this section. The seismic qualification criteria applicable to the seismic Category I equipment, and supports are provided. Methods used to qualify seismic Category I mechanical and electrical equipment, instrumentation, and supports are also provided.

The environmental qualification of the equipment including qualified life, and maintenance and surveillance programs are discussed in Section 3.11. The operability of active pumps and valves is discussed in Sections 3.9.3.2.2 and 3.9.3.2.3, respectively. Seismic qualification for Nuclear Steam Supply System (NSSS) equipment is discussed in Section 3.10N.

3.10.1 Seismic Qualification Criteria

The South Texas Project Electric Generating Station (STPEGS) design criteria meet the general requirements of the seismic qualification of seismic Category I equipment.

The seismic qualification and documentation procedures used for safety-related equipment and their supports meet the intent of Institute of Electrical and Electronic Engineers (IEEE) Standard 344-1975 and Regulatory Guide (RG) 1.100. The project compliance to RG 1.48 is noted in Section 3.12 and Table 3.9-2.5.

Seismic qualification of equipment by analysis and/or tests demonstrates that the equipment is able to withstand seismic loads as a result of the Safe Shutdown Earthquake (SSE) preceded by five Operating Basis Earthquakes (OBE) without loss of function in the operating mode.

The acceptance criteria for qualification of seismic Category I mechanical and electrical equipment and instrumentation are specified.

The functional operability criteria such as the operability of equipment during and/or after the SSE, and/or maintaining pressure integrity are specified.

The requirements for designing seismic Category I mechanical equipment that are qualified to maintain the pressure boundary integrity are in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III.

Equipment that has been previously qualified using methodologies equivalent to those described herein are acceptable provided that proper documentation is submitted and the loads and load combinations used in qualification envelop the project criteria.

Chapter 13.7 describes alternate requirements for safety related non-risk significant and low safety significant structures, components, and systems. These alternate requirements may be applied to both BOP and NSSS equipment.

3.10.1.1 Functional Monitoring. Seismic Category I mechanical and electrical equipment and instrumentation are qualified to demonstrate their operability.

To demonstrate proper functional operability, the normal mode of operation has been monitored during and after the seismic simulation or after the seismic simulation, whichever is applicable.

Monitoring equipment is required to monitor both input and output of the test specimen. The records confirm that the specimen performs all its safety-related functions within its “allowable” tolerance.

3.10.2 Methods for Qualifying Electrical Equipment and Instrumentation

3.10.2.1 Means of Qualification. IEEE Standard 344-1975 and RG 1.100 are used for seismic qualifications.

The horizontal and vertical OBE and SSE required response spectra (RRS) curves, as discussed in Section 3.7, form the basis for the seismic qualification of equipment, systems, and components. The RRS curves are identified with the building elevation and are a part of the specification, along with the acceptance criteria for the safety-related functions for each item of equipment.

The seismic qualification reports demonstrate (in accordance with Section 3.10.1) that the equipment performs its required safety-related function before, during, and after (as required) five OBEs followed by one SSE. For components that have been previously tested to generic criteria, test inputs are reviewed to assure that the test response spectra (TRS) envelops the applicable RRS over the frequency range of interest. Test reports are reviewed to confirm the required operability.

For active mechanical equipment (i.e., pumps and valves) test and/or analysis is used to demonstrate operability and structural integrity of components. Other seismic Category I safety-related mechanical equipment is qualified by analysis to demonstrate structural integrity. Load combinations, combining of dynamic responses for mechanical equipment, and the pump and valve operability assurance program are discussed in Section 3.9.

3.10.2.2 Method of Qualification. The methods for seismic qualification are listed blow:

- Analysis
- Test
- Combination of analysis and test

3.10.2.2.1 Analysis: Mathematical analyses without testing are acceptable if the structural integrity alone ensures the intended design function of the equipment (see Section 3.10.1) or if testing is impractical because of the size and weight of the equipment. The methodology used is in accordance with Section 5 of IEEE 344-1975.

When an equivalent static coefficient analysis is performed, justification for its use is provided. See Section 3.7.3A.1.2 for additional information on use of equivalent static load method of analysis.

Analytical results are evaluated as applicable for mechanical strength, fatigue, alignment, and noninterruption of function as related to the functional requirements of the equipment during an SSE event. Maximum stresses under all loading conditions are computed and compared with the allowables. Interference effects as well as interaction effects are considered in the analysis when significant.

3.10.2.2.2 Testing: Seismic tests are performed by subjecting equipment to vibratory motion that conservatively simulate the seismic vibratory environment at the equipment mounting location.

Seismic qualification by testing is performed using either multifrequency or single frequency inputs. These test inputs and methods are in accordance with IEEE 344-1975, Section 6.

The multifrequency test method is used for floor- and wall-mounted equipment. In addition, in special cases it is used for equipment mounted on structural steel, piping, ducts, or other types of supports or equipment where an analysis or test has been performed to determine the RRS at the equipment mounting location. These tests or analyses consider the dynamic amplification characteristics of the support system.

For equipment qualified by multifrequency testing the measured TRS envelopes the RRS in the frequency range of interest.

Single-frequency tests are used for line-mounted equipment, which includes equipment mounted in piping systems and in ducts. The equipment is tested to a required input motion (RIM). The RIM is the peak acceleration of the input motion (sine wave or sine beats) at a specified frequency. The piping and duct systems are designed and supported to limit the peak acceleration experienced by the equipment to a value less than the specified RIM acceleration.

Single-frequency tests may also be used for other types of equipment as permitted by IEEE 344-1975 and RG 1.100.

3.10.2.2.3 Combined Test and Analysis: When equipment cannot be qualified practically by analysis or testing because of its complexity, size, or weight, combined analysis and testing is utilized. This method of qualification is applied to equipment such as cabinets that may contain several different configurations of internally mounted devices.

The combined analysis and test method is in accordance with Section 7 of IEEE 344-1975, and the equipment qualification methods of Sections 3.10.2.2.1 and 3.10.2.2.2.

Equipment that has been previously qualified by means of test and analysis equivalent to those described herein is acceptable if proper documentation is provided.

3.10.2.3 Test Sequence Verification. As defined in Part B of RG 1.100, IEEE 344-1975 is an ancillary standard of IEEE 323-1974. In accordance with this standard, seismic testing as part of the overall qualification is performed in its proper sequence as indicated in Section 6 of IEEE 323-1974.

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Mechanical and Electrical Equipment and Instrumentation

Analysis and/or test is performed for seismic Category I equipment supports to ensure their structural capability to withstand seismic excitation.

Information concerning the structural integrity of pressure-retaining components, their supports, and core supports is presented in Section 3.9.3. The following bases are used in the analysis and design of cable tray supports, heating, ventilating and air conditioning (HVAC) ducts supports and instrument tubing supports.

1. The methods used in the seismic analysis of cable tray and HVAC duct supports are described in Section 3.7.3A.1.2 and 3.7.3A.15. The amplification of seismic loads due to the flexibility of the supporting system, if any, is accounted for in the design of the cable trays and in the qualification of duct mounted equipment.

2. The seismic Category I instrument tubing systems are supported so that the allowable stresses permitted by Section III of ASME B&PV Code are not exceeded when the tubing is subjected to the loads specified in Section 3.9 for Class 2 and 3 piping.

For field-mounted instruments the supports are tested or analyzed to meet the following:

1. The field mounting supports for seismic Category I instruments excluding line-mounted instruments have a fundamental frequency of 33 Hz or greater, with the weight of the instrument included. If, however, the mounting should be flexible (i.e., frequency <33 Hz), the dynamics of the support are considered in the qualification of the supported instrument.

2. The stress level in the mounting support does not exceed the material allowable stress when subjected to the maximum acceleration level of the mounting location. The weight of the instrument is included. In some cases, panels and racks supporting seismic Category I devices are tested and/or analyzed with equipment installed. If the devices are in an inoperative mode during the support test, the response at the device mounting location is monitored. In such a case, devices are qualified separately, and the actual input to the devices is more conservative in amplitude and frequency content than the response monitored at the devices' location. The RRS for device (i.e., in-cabinet response spectra) are generated and, as shown in the individual qualification reports as applicable to the device and the test response spectra to which the device is qualified, envelops the RRS measured at the device mounting location.

3.10.4 Operating License Review

The method of qualification and the results are identified in the Seismic Master List submittal. Equipment qualification documentation is stored in the Record Management System (RMS) in a retrievable and auditable form. This documentation will be available for the life of the plant. The documentation procedures for safety-related seismic Category I equipment are in accordance with the recommendations contained in IEEE 344-1975 and RG 1.100.

3.10N SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT (NSSS)

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 an earthquake. The information presented includes identification of the Category I instrumentation and electrical equipment that is within the scope of the Westinghouse Nuclear Steam Supply System (NSSS) and the qualification criteria employed. Included 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.

The environmental qualification of the equipment including qualified life is described in Section 3.11N. The operability of active pumps and valves is discussed in Section 3.9.3.2.1.

3.10N.1 Seismic qualification Criteria

3.10N.1.1 Qualification Standards. The Nuclear Regulatory Commission's (NRC) recommendations concerning the methods to be employed for seismic qualification of electrical equipment are contained in Regulatory Guide (RG) 1.100, which endorses Institute of Electrical and Electronic Engineers (IEEE) 344-1975. Westinghouse meets this standard, as modified by RG 1.100, by either type test, analysis, or an appropriate combination of these methods. Westinghouse meets this commitment employing the methodology described in Reference 3.10N-1.

According to RG 1.89, equipment for plants in the stage of construction permit application and having the issue data of the Safety Evaluation Report after July 1, 1974 take into account aging and environmental effects prior to seismic qualification, as specified in the IEEE Standard 323-1974. Westinghouse meets IEEE 323-1974 by either type test, analysis, or an appropriate combination of these methods. Required seismic tests conform to the procedures specified in IEEE 344-1975 which account for multiaxis and multifrequency effects of seismic excitation and fatigue effects caused by a number of DBE events. Westinghouse meets these commitments by employing the methodology described in WCAP-8587 (Ref. 3.10N-1). This WCAP was reviewed and accepted by the NRC through the issuance of a Safety Evaluation Report (SER) on November 10, 1983.

Reference 3.10N-2 presents the Westinghouse testing procedure used to qualify equipment by type testing. Seismic qualification testing of equipment to IEEE 344-1971 is documented in References 3.10N-3 through 3.10N-8. Reference 3.10N-9 presents the theory, practice, and justification for use of single axis sine beats test inputs used in seismic qualification. In addition, it is noted that Westinghouse has conducted a seismic qualification "Demonstration Test Program" Reference 3.10N-10 to confirm equipment operability during a seismic event.

For the seismic qualification of Westinghouse electrical equipment outside of the containment, the above-noted demonstration test program, in conjunction with the justification for the use of single axis sine beat tests (presented in Reference 3.10N-13) and the original tests (documents in Reference 3.10N-3 through 3.10N-8, 3.10N-13) meets the requirements of IEEE Standard 344-1975.

Thus, since the "Demonstration Test Program" was successfully completed, the equipment's operability has been demonstrated to the requirements of IEEE Standard 344-1975.

The acceptability criteria for the SSE notes that there may be permanent deformation of the equipment provided that the capability to perform its function is maintained.

3.10N.1.2 Performance Requirements for Seismic Qualification. Reference 3.10N-11 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 3.10N-1 identifies the Category I equipment supplied by Westinghouse for this application and references the applicable EQDP contained in Reference 310N-11. Each EQDP contains a section entitled "Performance Requirements". 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. These generic seismic spectra envelope South Texas Project Electric Generating Station (STPEGS) specific floor response spectra for the equipment location.

3.10N.1.3 Acceptance Criteria. Seismic qualification must demonstrate that 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.

3.10N.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 EQDPs of Reference 3.10N-11.

3.10N.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 3.10N-2. In addition, Westinghouse has conducted seismic retesting of certain items of equipment as part of the Supplemental Qualification Program (Ref. 3.10N-10).

This retesting was performed at the request of the NRC staff on agreed selected items of equipment employing multi-frequency, multi-axis tests inputs (Ref. 3.10N-12) 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 NRC. 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

(Ref. 3.10N-10), no additional qualification testing is required to demonstrate acceptability to IEEE 344-1975 provided that:

1. 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.

2. Any changes made to the equipment due to item 1, above, or due to design modifications do not significantly affect the seismic characteristics of the equipment.

3. 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 3.10N-1, Table 7.1 and the test results in the applicable EQDPs of Reference 3.10N-11.

For equipment tests after July 1974 (i.e., new designs, equipment not previously qualified or previously qualified that does not meet 1, 2 and 3 above) seismic qualification by test is performed in accordance with IEEE 344-1975. Where testing is utilized, multi-frequency multi-axis inputs are developed by the general procedures outlined in Reference 3.10N-14. The test results contained in the individual EQDPs of Reference 3.10N-11 demonstrate that the measured Test Response Spectrum (TRS) envelopes the applicable RRS defined for generic testing as specified in Section 1.8 of each EQDP (Ref. 3.10N-11). 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, single axis inputs, are used in selected cases as permitted by IEEE 344-1975 and RG 1.100.

3.10N.2.2 Seismic Qualification by Analysis. Employing motors as an example, the structural integrity of safety-related motors 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, 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 (Ref. 3.10N-11).

3.10N.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 3.10N-11 identify the equipment mounting employed for qualification purposes and establish interface requirements for the equipment to ensure subsequent in-plant installation does not prejudice the qualification established by Westinghouse.

3.10N.4 Operating License Review

The results of tests and analyses that ensure that the criteria established in Section 3.10N.1 have been satisfied employing the qualification methods described in Section 3.10N.2 and 3.10N.3 are included in the individual EQDPs contained in Reference 3.10N-11.

REFERENCES

Section 3.10N:

3.10N-1 WCAP-8587, Revision 6-A (Nonproprietary) 1983, G Butterworth and R.B. Miller, "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment".

3.10N-2 WCAP-7558 (Nonproprietary) 1971, A. Morrone, "Seismic Vibration Testing with Sine Beats".

3.10N-3 WCAP-7397-L (Proprietary) January 1970 and WCAP-7817 (Nonproprietary), 1971, E.L. Vogeding, "Seismic Testing of Electrical and Control Equipment".

3.10N-4 WCAP-7397-L, Supplement 1 (Proprietary) January 1971 and WCAP-7871, Supplement 1 (Nonproprietary), December 1971, E.L. Vogeding, "Seismic Testing of Electrical and Control Equipment (WCID Process Control Equipment)".

3.10N-5 WCAP-7817, Supplement 2 (Nonproprietary) 1971, L. M. Potochnik, "Seismic Testing of Electric and Control Equipment (Low Seismic Plants)".

3.10N-6 WCAP-7871, Supplement 3 (Nonproprietary) 1971, E.L. Vogeding, "Seismic Testing of Electric and Control Equipment (Westinghouse Solid State Protection System) (Low Seismic Plants)".

3.10N-7 WCAP-7817, Supplement 4 (Nonproprietary) 1972, J.B. Reid, "Seismic Testing of Electrical and Control Equipment (WCID NUCANA 7300 Series) (Low Seismic Plants)".

3.10N-8 WCAP-7817, Supplement 5 (Nonproprietary), E.L. Vodeging, "Seismic Testing of Electrical and Control Equipment (Instrument Bus Distribution Panel)".

3.10N-9 WCAP-8373 (Nonproprietary), E.G. Fischer and S.J. Jarecki, "Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May 1984".

3.10N-10 Eicheldinger, C., Westinghouse July 10, 1975, letter NS-CS-692 to D.B. Vassallo, NRC.

3.10N-11 WCAP-8587, Supplement 1 (Nonproprietary), EQDP – "Equipment Qualification Data Packages".

3.10N-12 WCAP-8624 (Proprietary), S.J. Jarecki, "General Method of Developing Multi-Frequency Biaxial Test Inputs for Bistables".

REFERENCES (Continued)

Section 3.10N:

3.10N-13 WCAP-7817, Supplement 6 (Proprietary) 1974, E.K. Figenbaum and E.L. Vogeding, "Seismic Testing of Electrical and Control Equipment (Type DB Reactor Trip Switchgear)".

3.10N-14 WCAP-9714-PA (Proprietary), WCAP-9750-A (Nonproprietary), May 1980, R.E. Kelly, and J.J. McInerney, "Methodology for the Seismic Qualification of Westinghouse WRP Supplied Equipment".

3.10N-15 WCAP-8687 Appendix A1, Revision 1 (Proprietary) R. Jabs, et. al., "Short Term Component Aging Test Program".

3.10N-16 WCAP-8687 Appendix A5 (Proprietary) M.S. Chang and R. Jabs, "Microprocessors and Supporting Electronic Components Aging Test Program".

Table 3.10N-1

SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY (

Equipment	EQDP*
Medium Pump Motors (Outside Containment)	AE-1
Large Pump Motors (Outside Containment)	AE-2
Chempump Canned Motor Pump (Outside Containment)	AE-3
Veritrak Pressure Transmitters (Group A)	ESE-1B and Addendum
Tobar Pressure Transmitters (Group A)	ESE-1C
Barton & Veritrak Pressure Transmitters (Group B)	ESE-2
Tobar Pressure Transmitters (Group B)	ESE-2C
Rosemount Pressure Transmitters (Group B)	ESE-2D
Barton DP Transmitters (Group A)	ESE-3A
Veritrak DP Transmitters (Group A)	ESE-3B and Addendum
Tobar DP Transmitters (Group A)	ESE-3C
Barton DP Transmitters (Group B)	ESE-4
Tobar DP Transmitters (Group B)	ESE-4C
RdF TRDs RCS Well Mounted	ESE-6B
RdF RTDs – Fast Response Well Mounted	ESE-7B
2 Section Excore Neutron Detectors (Power Range) Bottom Inserted	ESE-8B

Table 3.10N-1 (Continued)

SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY

Equipment	EQDP*
Excore Neutron Detectors (Source and Intermediate Range)	ESE-9A
Nuclear Instrumentation System (NIS) Console (Power Range Channel)	ESE-10
Nuclear Instrumentation System (NIS) Source Range Pre-Amplifier (Excontainment)	ESE-11A
Operator Interface Modules (OIMs)	ESE-12A
Process Protection System	ESE-13
Indicators: Post-Accident Monitoring	ESE-14
Recorders: Post-Accident Monitoring	ESE-15
Three Train Solid State Protection System and Safeguards Test Cabinets	ESE-17
Instrument Bus Power Supply (Static Inverter)	ESE-18
Reactor Trip Switchgear	ESE-20
Pressure Sensor	ESE-21
Minco Surface Mounted RTDs	ESE-42A
Incore Thermocouples	ESE-43A
Thermocouple Connectors	ESE-43B
Thermocouple Connector for IHP Plants (Panel Mounted)	ESE-43D
Thermocouple Adapters and Cable Splice Assemblies	ESE-43G
Thermocouple Connectors Reference Junction Box	ESE-44A
Low Noise Source Range Pre-Amplifier	ESE-47B
Barton High Volume Sensor	ESE-48A
Barton D/P Indicating Switch	ESE-49A

Table 3.10N-1 (Continued)

SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY

Equipment	EQDP*
PSMS **	ESE-53
PSMS Components **	ESE-53B
Eagle 21 Components **	ESE-53C
PSMS Appendix R Interfaces **	ESE-53D
Auto Shunt Trip Panel and Attachment	ESE-62A
PSMS Plasma Display and Keyboard **	ESE-63A
PSMS Modular Plasma Display **	ESE-63B
Limitorque Valve Motor Operators (Qualification Group A with Class F; Type LR Insulation)	HE-1
ASCO Solenoid Valves	HE-2/ HE-5 and Addendum
Namco Externally Mounted Limit Switches	HE-3/ HE-6
Limitorque Valve Operators (Qualification Group B with Class B Insulation)	HE-4
Conax Electrical Connector Seal Assemblies (ECSAs) Valves and Limit Switches	HE-8
Garrett PORV Solenoid Operated Pilot Valve and Position Indication Device	HE-9
Head Vent System: Target Rock Solenoid Operated Isolation Valve	HE-10A
Head Vent System: Target Rock Electronic Control Module	HE-10B
Head Vent System: Target Rock Modulating Valve	HE-10C

Table 3.10N-1 (Continued)

SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL

EQUIPMENT IN WESTINGHOUSE NSSS SCOPE OF SUPPLY

CN-2867

Equipment	EQDP*
Chicago Fluid Power MSIV Solenoids	WCAP-11160
Limitorgue Valve Operators (with Class H, Type RH Motor Insulation)	WCAP-11172
RHR Pump Motors	WCAP-11216

* Equipment Qualification Data Package
** Associated with Qualified Display Processing System

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

Safety-related electrical equipment is designed to remain functional during and following design basis events. In addition, certain post-accident monitoring equipment, is also designed to remain functional during or after specified design basis events, or to not fail in a manner which could prevent satisfactory accomplishment of the plant safety functions.

Design basis events consist of normal operation and plant shutdown, loss of offsite power (LOOP) and design basis accidents (DBA).

The following sections provide information to demonstrate acceptable performance of Non-Nuclear Steam Supply System (NSSS) (i.e., balance of plant) equipment under the specified conditions. Environmental qualification for NSSS equipment is discussed in Section 3.11N.

The programs for preventive maintenance, surveillance and periodic testing have been developed in accordance with Regulatory Guide (RG) 1.33, Rev. 2. These programs are based on manufacturer recommendations, experience and the results of the project qualification programs. This will ensure that all safety-related equipment in mild and harsh areas will be operable and qualified throughout the life of the plant.

The programs provide for replacement of parts and equipment prior to the end of qualified life.

Chapter 13.7 describes alternate requirements for safety related non-risk significant and low safety significant structures, components, and systems. These alternate requirements may be applied to both BOP and NSSS equipment.

3.11.1 Equipment Identification and Environmental Conditions

A complete list of safety-related electrical equipment required to be qualified was provided in the 10CFR50.49 submittal and is maintained in accordance with plant procedures. A list of all Category 1 and 2 post-accident monitoring equipment (in response to RG 1.97, Rev. 2) that is included in the equipment qualification program is provided in Table 7.5-1. Worst case expected environmental conditions for each area in which the subject equipment is installed are listed in Table 3.11-1. The conditions are based on the following:

1. Normal parameters are those which will be maintained during routine plant operation, shutdown, hot standby, and system testing. The range is based on the limiting conditions of peak outdoor temperature together with equipment design heat loads and minimum outdoor temperature together with no heat loads.

2. Abnormal parameters are those which may be caused by such events as loss of nonsafety-related heating ventilation, and air conditioning (HVAC). The majority of qualified equipment areas are served by safety class HVAC, for which outages due to LOOP are not postulated.

3. Accident conditions are those plant conditions resulting from the most limiting pipe failure for that location during which safety-related equipment must operate to mitigate the consequences of the accident.

4. Normal cumulative radiation doses are the totals projected for a 40-year plant life, utilizing the shielding assumptions of Section 12.3.2. Accident cumulative radiation doses are based on the source terms of Section 3.11.5.2 and are totaled through the period of 180 days past initiation of the accident.

For pipe or line mounted electrical or electro-mechanical devices, the plant vibration effects are considered in accordance with the guidelines provided in Institute of Electrical and Electronic Engineers (IEEE) 382-1972. For floor and wall mounted equipment, the simulation of five Operating Basis Earthquakes (OBE) before a Safe Shutdown Earthquake (SSE) are considered to include the vibration aging effects on equipment.

3.11.2 Qualification Tests and Analyses

Safety-related electrical equipment and components located in a harsh environment are qualified by test or combination of test and analysis in accordance with the requirements of 10CFR50.49 and NUREG-0588, Rev. 1. Qualification testing is accomplished either by tests on the particular equipment or by type tests performed on similar equipment under environmental conditions at least as severe as the specified conditions.

For safety-related mechanical equipment, compliance with the GDC-4 requirements is established through the evaluation of non-metallic parts based on the FIT, FORM and FUNCTION methodology used in the Item Equivalency Evaluation and the Commercial Grade Dedication process of the STP Procurement Program.

For safety-related electrical equipment located in a mild environment, the design/purchase specifications provide the documentation required to demonstrate qualification. The maintenance and surveillance program in conjunction with a trending program provide the assurance that equipment which meets the design/purchase specifications is maintained throughout its life. Documented operating experience may be utilized for qualification of equipment located in a mild environment.

Project compliance with environmental qualification criteria can be found in the following sections:

		Sections
10CFR50, Appendix A	GDC 1	3.1.2.1
	GDC 4	3.1.2.1
	GDC 23	3.1.2.3
	GDC 50	3.1.2.5
10CFR50, Appendix B		17.2
USNRC Regulatory Guides	RG 1.30	3.12
	RG 1.33	3.12
	RG 1.40	3.12
	RG 1.63	3.12
	RG 1.73	3.12

RG 1.89	3.12
RG 1.97	3.12
RG 1.131	3.12

3.11.2.1 Qualification Requirements to Vendor. The following information and requirements were specified in equipment purchase specifications.

1. Vendors have been required to submit a description of the method of qualification performed on each specified safety-related item located in the Containment and elsewhere to assure it will perform satisfactorily in the normal, abnormal and accident environment of temperature, pressure, humidity, chemical, and radiation doses.

2. Vendors have been required to provide evidence concerning the satisfactory behavior of proposed materials under the environmental conditions specified. Data on changes in material properties have been evaluated for adequacy.

Acceptable qualification programs, at the minimum, demonstrate the end-of-life qualification. The methods employed for meeting the IEEE 323-1974 aging requirements for safety-related equipment are documented in the individual equipment qualification data package and vendor qualification reports.

Qualification programs which do not demonstrate the qualification of equipment for its specified period of design life are identified with a supporting maintenance, replacement, and surveillance program. Acceptable qualification programs include prototype tests and/or analysis under conditions simulating the environmental conditions expected over the 40-year life plus the 30 days post-accident period for temperature and pressure and l80 days post-accident period for radiation in accordance with standards listed in Tables 3.11-3 and 3.11-4.

The conditions imposed for test and/or analysis include normal, abnormal, and DBA environmental conditions postulated to occur during the period of life for which the equipment is qualified.

Class 1E cables, field splices, and terminations for use on the South Texas Project Electric Generating Station (STPEGS) with the exception of single conductor high temperature silicon insulated cables meet the requirements of IEEE 383-1974 as modified by RG 1.131. Single conductor high temperature silicon insulated cables when used in a class 1E circuit are installed in conduit only. Polyethylene cables used in safety-related applications are identified in Table 3.11-5.

3.11.3 Qualification Test Results

Detailed qualification results for electrical equipment located in a harsh environment appears in the station files. Plant specific evaluations are performed to ensure that the generic testing performed by vendors encompasses the plant specific environmental conditions. The qualified life of equipment is extended or reduced based on specific plant variables such as environmental parameters, operational cycles, performance characteristics and properties of the materials used in construction of the equipment.

3.11.4 Loss of Ventilation

The majority of qualified equipment areas are served by safety class HVAC. These HVAC systems are designed to the single failure criteria and are supplied from the Onsite Standby Power System. Consequently, the normal environmental conditions which they provide will be maintained during all plant modes. However, certain areas of the plant served by safety class HVAC may experience abnormal temperature conditions due to loss of offsite power which would result from switchover to a different cooling medium.

A small amount of qualified equipment is in areas served only by nonsafety HVAC. For these areas, the abnormal ranges of environmental conditions are based on the loss of HVAC.

Table 3.11-1 provides a listing of the worst-case environmental conditions for various areas in the plant. These conditions were determined by the criteria listed in Section 3.11.1.

3.11.5 Estimated Chemical and Radiation Environment

3.11.5.1 Chemical Environment. Safety-related systems and components were originally designed to perform their functions on long-term contact with acidic and basic solutions recirculated through the Emergency Core Cooling System and Containment Spray System following a LOCA. An initial pH of 4.5 is due to the addition of 2000-4000 ppm boric acid. Thereafter, pH ranges from 7.0-9.5 due to addition of TSP. Spray flow is vertical and ranges from 0 to 0.5 gpm/ft^2. Spray duration is 24 hrs in accordance with IEEE 323-1974 for the purpose of equipment qualification.

A plant modification has resulted in a change to the containment spray pH and the sump pH as described in Section 6.5.2. The new chemical environment is based on the deletion of the additive (sodium hydroxide) to the containment spray and the use of trisodium phosphate (TSP) to adjust the sump pH.

A review has been done to show the materials for equipment which have been qualified for the original design condition of NaOH spray of 7.5 to 10.5 pH are either not affected by the change to the new pH environment or replaced with a suitable material if affected.

3.11.5.2 Radiation Environment. Safety-related systems and components are designed to perform their safety-related functions after normal operation radiation exposure plus a DBA exposure. The normal operational exposure is based on the design basis source terms presented in Sections 11.1, 11.2, 11.3, and 12.2.1 and the equipment and shielding configurations given in Section 12.3.

The effect of the Vantage 5H (V5H) fuel upgrade on radioactivity concentrations in the fluid systems was reviewed and it was determined that the original reactor coolant activity listed in Table 11.1-2 is bounding. Therefore, the FSAR analyses based on this activity are not adversely impacted by the fuel upgrade. For comparison, the activity concentrations calculated for the V5H fuel are listed in Table 11.1-2A. The corresponding reactor core activity for the V5H upgrade is shown in Table 15.A-1A.

Safety-related system and component radiation exposures are dependent on equipment location and the particular DBA involved. In the Containment and control room area, equipment exposures are based on the DBA LOCA. For in-Containment equipment, the DBA LOCA source term is based on a release of 100 percent of the core noble gases, 50 percent of the halogens and 1 percent of the solids. This is consistent with the guidance given in RG 1.89. Control room exposures following a postulated LOCA are controlled to 5 rads or less consistent with the requirements of GDC 19 of 10CFR50, Appendix A. The source terms used correspond to a cycle length of approximately 20,000 MWD/MTU, a core average burnup of 40,000 MWD/MTU, and a discharge burnup of 60,000 MWD/MTU. These burnups are conservative relative to the planned cycle lengths for V5H fuel described in Section 4.3.

Radiation source terms for safety-related components which are exposed to post-accident recirculation fluid are consistent with the recommendations of RG 1.89 (i.e., 50 percent of the core halogen inventory and 1 percent of the remaining core solid fission product inventory are mixed in the recirculation water).

Normal and accident radiation doses for the various plant areas are presented in Table 3.11-1. Safety-related equipment design doses are the sum of normal plus accident exposures. The design radiation exposures delineated in Table 3.11-1 are based on gamma and beta radiation. Radiation source terms for safety-related components outside Containment are based on gamma radiation.

Organic materials in the Containment are identified in Section 6.l.2. For the organic coating materials used inside Containment (see Section 6.l.2.l), irradiation tests performed by Oak Ridge National Laboratory have been performed for an integrated gamma dose of 1 x 10^9 rads (which exceeds the design calculated value in Table 3.11-1). These doses conservatively account for the surface exposure due to beta radiation in the design basis LOCA environment.

TABLE 3.11-1

ENVIRONMENTAL CONDITIONS

Location	Temperature			Pressure		Relative Humidity		Cumulative Radiation[2] Dosage		
(Environmental Designator)	Normal Range (max/min. °F)	Abnormal	Accident (°F)	Normal	Accident	Normal Range (max/min. %)	Accident (%)	Normal (rads)	Accident (rads)	Radiation Type
1. Reactor Containment[1,3]										
Inside CRDM Shroud	200/65	226/-	299	+0.3/psig max -0.1 psig min	41.2 psig max -3.1 psig min	70/0	100	$7x10^5$	$1.2x10^8$	gamma and beta
Steam Generator Compartments (Rm. 201)	120/65	159/-	299	+0.3/psig max -0.1 psig min	41.2 psig max -3.1 psig min	70/0	100	$2x10^7$	$1.2x10^8$	gamma and beta
Pressurizer Enclosure (Rm. 206)	150/65	197/-	299	+0.3/psig max -0.1 psig min	41.2 psig max -3.1 psig min	70/0	100	$2x10^7$	$1.2x10^8$	gamma and beta

1. LOCA chemistry conditions are specified in Section 3.11.5.1
2. Radiation exposure values represent the 40-year Integrated exposures for normal operation and 180 days post-accident values.
3. Inside Containment normal radiation parameters are gamma only, while accident parameters are gamma and beta.

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITIONS

Location (Environmental Designator)	Temperature Normal Range (max/min. °F)	Temperature Abnormal (max/min. °F)	Temperature Accident (°F)	Pressure Normal	Pressure Accident	Relative Humidity Normal Range (max/min. %)	Relative Humidity Accident (%)	Cumulative Radiation[2] Dosage Normal (rads)	Cumulative Radiation[2] Dosage Accident (rads)	Radiation Type
Reactor Cavity (Rms. 001, 002)	135/65	142/-	299	+0.3/psig max -0.1 psig min	41.2 psig max -3.1 psig min	70/0	100	$3.5x10^9$	$1.5x10^8$ $2.5x10^{10}$	gamma beta neutron
Other Areas Inside Secondary Shield (Below El. 19 ft, Rm. 004)	120/65	167/-	299	+0.3/psig max -0.1 psig min	41.2 psig max -3.1 psig min	70/0	100	$3.5x10^5$	$1.5x10^8$	gamma and beta
Other Areas Inside Secondary Shield (above El. 19 ft)	120/65	165/-	299	+0.3 psig max -0.1 psig min	41.2 psig max -3.1 psig min	70/0	100	$2x10^7$	$1.2x10^8$	gamma and beta
Other Areas Outside Secondary Shield	120/65	168/-	299	+0.3/psig max -0.1 psig min	41.2 psig max -3.1 psig min	70/0	100	$3.5x10^4$	$1.5x10^8$	gamma and beta
RHR Pump and Heat Exchanger Rooms; Valve Rooms (Rms. 104, 109, 110, 303, 304, 306, 105, 108, 111, 202, 209, 207)	120/65	167/-	299	+0.3/psig max -0.1 psig min	41.2 psig max -3.1 psig min	70/0	100	$7x10^5$	$1.2x10^8$	gamma and beta
Tendon Access Gallery (Rms. 011, 013)	95/50	120	NA	atm.	atm.	80/20	100	100	100	gamma

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITIONS

Location (Environmental Designator)	Temperature Normal Range (max/min. °F)	Temperature Abnormal (max/min. °F)	Temperature Accident (°F)	Pressure Normal	Pressure Accident	Relative Humidity Normal Range (max/min. %)	Relative Humidity Accident (%)	Cumulative Radiation[(2)] Dosage Normal (rads)	Cumulative Radiation[(2)] Dosage Accident (rads)	Radiation Type
2. Isolation Valve Cubicles and Penetration Areas										
Watertight Rooms (Rms. 001, 002, 003, 004)	129/40	129/40	225	atm.	4.5 psig	80/20	100	100	100	gamma
Aux. Feedwater Pump Cubicles (Rms. 005, 006, 007, 008) and Platforms, El. 21 ft 2 in. (Rms. 101, 102, 103, 104)	104/50	104/40	335	slightly positive	5.8 psig	80/20	100	100	$1.3x10^4$	gamma
Feedwater Cubicles (Rms. 201, 202, 203 204) and Feedwater Line Area (Rms. 301, 302, 303, 304)	104/50	104/36	335	slightly positive	5.8 psig	80/20	100	100	$1.6x10^5$	gamma
Steam line Area (Rms. 401, 402, 403, 404)	104/50	104/36	335	slightly positive	5.8 psig	80/20	100	100	$3.5x10^5$	gamma
(Rms. 501, 502, 503, 504)	104/50	104/36	335	slightly positive	2.8 psig	80/20	100	100	$3.5x10^5$	gamma
3. Electrical Auxiliary Building										
Electrical Penetration Room (Rm. 001)	104/50	104/50	104	atm.	atm.	80/20	80	100	$8.5x10^5$+	gamma

+ Neutron Flux Amplifier Channel A Panel ZLP-685 has an accident dose of less than 10^5 rads (NC9040).

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location (Environmental Designator)	Temperature Normal Range (max/min. °F)	Temperature Abnormal (max/min. °F)	Temperature Accident (°F)	Pressure Normal	Pressure Accident	Relative Humidity Normal Range (max/min. %)	Relative Humidity Accident (%)	Cumulative Radiation(2) Dosage Normal (rads)	Cumulative Radiation(2) Dosage Accident (rads)	Radiation Type
Power Cable Vault (Rm. 002)	104/50	104/50	104	atm.	atm.	80/20	80	100	$1.3x10^3$	gamma
Entrance at Elect. Penetration (Rm. 003)	104/50**	104/50**	104	atm.	atm.	80/20	80	100	$8.5x10^3$	gamma
Corridor (Rm. 004)	104/50	104/50	104	atm.	atm.	80/20	80	100	$1.3x10^3$	gamma
Fire Extinguish-ing System Room (Rm. 005)	104/50	104/50	104	atm.	atm.	80/20	80	100	$1.3x10^3$	gamma
Battery Room (Rms. 006, 008)	77/65	77/65	77	slightly negative	slightly negative	75/45	80	100	$1.3x10^3$	gamma
Distribution Panel (Rms. 007, 009)	104/50	104/50	104	atm.	atm.	80/20	80	100	100	gamma
Electrical Equipment Room (Rm. 008A)	104/50	104/50	104	atm.	atm.	80/20	80	100	100	gamma
Switchgear Room (Rm. 010)	86/50	86/50	104	atm.	atm.	80/20	80	100	100	gamma
HVAC Room (Rm. 013)	104/50	104/50	104	slightly positive	+0.125" wg.	80/20	80	100	500	gamma
Aux. Shutdown Panel (Rms. 015, B, C, D)	80/50	80/50	80	atm.	atm.	80/20	80	100	100	gamma
Electrical Cable Room (Rm. 101)	104/50	104/50	104	atm.	atm.	80/20	80	100	$1.3x10^3$	gamma

** There is no HVAC or equipment in this area; therefore, the temperatures listed are based upon the adjacent room temperatures.

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location	Temperature			Pressure		Relative Humidity		Cumulative Radiation(2) Dosage		
(Environmental Designator)	Normal Range (max/min. °F)	Abnormal (max/min. °F)	Accident (°F)	Normal	Accident	Normal Range (max/min. %)	Accident (%)	Normal (rads)	Accident (rads)	Radiation Type
Cable Spreading Room (Rm. 102)	104/50	104/50	104	atm.	atm.	80/20	80	100	1.3×10^3	gamma
HVAC Room (Rm. 104B)	104/50	104/50	104	atm.	atm.	80/20	80	100	500	gamma
Electrical Penetration Area (Rm. 201)	104/50	104/50	104	atm.	atm.	80/20	80	100	5.1×10^5	gamma
Relay Room (Rm. 202)	80/50	80/50	80	slightly positive	+0.125"wg	80/20	80	100	100	gamma
Control Room (Rm. 203)	78/72	78/50	78	slightly positive	+0.125"wg	80/20	80	100	100	gamma
Results Eng. Office (Rm. 203B)	78/72	78/50	78	slightly positive	+0.125"wg.	80/20	80	100	100	gamma
Kitchen (Rm. 205C)	78/72	78/50	78	slightly positive	+0.125" wg.	80/20	80	100	500	gamma
HVAC Room (Rm. 206)	104/50	104/50	104	slightly positive	+0.125" wg.	80/20	80	100	500	gamma
HVAC Room (Rm. 206B)	104/50	104/50	104	atm.	atm.	80/20	80	100	500	gamma
Switchgear Room (Rm. 212)	86/50	86/50	104	atm.	atm.	80/20	80	100	130	gamma
Distribution Room (Rms. 213, 215B)	104/50	104/50	104	atm.	atm.	80/20	80	100	130	gamma
Battery Room (Rm. 214)	77/65	77/65	77	slightly negative	slightly negative	75/45	80	100	130	gamma

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location (Environmental Designator)	Temperature Normal Range (max/min. °F)	Temperature Abnormal (max/min. °F)	Temperature Accident (°F)	Pressure Normal	Pressure Accident	Relative Humidity Normal Range (max/min. %)	Relative Humidity Accident (%)	Cumulative Radiation[(2)] Dosage Normal (rads)	Cumulative Radiation[(2)] Dosage Accident (rads)	Radiation Type
Computer Room (Rm. 215)	72/68	72/68	72	slightly positive	+0.125"wg	60/40	60	100	130	gamma
Electrical Penetration Area (Rm. 301)	104/50	104/50	104	atm.	atm.	80/20	80	100	$7x10^5$++	gamma
Cable Spreading Room (Rm. 302)	104/50	104/50	104	atm.	atm.	80/20	80	100	$4x10^3$	gamma
HVAC Room (Rm. 307)	104/50	104/50	104	slightly positive	+0.125"wg.	80/20	80	100	500	gamma
Corridor gamma (Rm. 308)	78/72	78/50	78	atm.	atm.	80/20	80	100	130	gamma
Corridor (Rm. 308A)	104/50	104/50	104	atm.	atm.	80/20	80	100	$3x10^3$	gamma
Switchgear Room (Rm. 318)	86/50	86/50	104	atm.	atm.	80/20	80	100	100	gamma
Distribution Room Area (Rm. 319)	104/50	104/50	104	atm.	atm.	80/20	80	100	$1.3x10^3$	gamma
Motor Generator Set Room (Rm. 320)	104/50	104/50	104	atm.	atm.	80/20	80	100	$1.3x10^3$	gamma
Battery Room (Rm. 321)	77/65	77/65	77	slightly negative	slightly negative	75/45	80	100	$1.3x10^3$	gamma
Power Cabinets (Rm. 323)	104/50	104/50	104	atm.	atm.	80/20	80	100	$4x10^3$	gamma

++ Neutron Flux Amplifier Channel C Panel ZLP-686 has an accident dose of less than 10^5 rads (NC9040).

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location (Environmental Designator)	Temperature Normal Range (max/min. °F)	Temperature Abnormal (max/min. °F)	Temperature Accident (°F)	Pressure Normal	Pressure Accident	Relative Humidity Normal Range (max/min. %)	Relative Humidity Accident (%)	Cumulative Radiation[2] Dosage Normal (rads)	Cumulative Radiation[2] Dosage Accident (rads)	Radiation Type
Power Cable Vault (Rm. 401)	104/50	104/50	104	atm.	atm.	80/20	80	100	$4x10^3$	gamma
Computer Room (Rm. 402)	77/73	77/73	NA	slightly positive	at least 1/8-in. positive	40	NA	100	100	gamma
Storage Room (Rm. 403)	78/72	78/72	78	atm.	atm.	80/20	80	100	100	gamma
HVAC Room (Rm. 410)	104/50	104/50	104	atm.	atm.	80/20	80	100	500	gamma
Cable Vault Room (Rm. 412)	104/50	104/50	104	atm.	atm.	80/20	80	100	$4x10^3$	gamma
Outside Air Intake Structure (Rm. 501)	104/29	104/29	104	atm.	atm.	80/20	100	100	$1.3x10^4$	gamma
HVAC Room (Rm. 502B)	104/50	104/50	104	atm.	atm.	80/20	80	100	$1.3x10^4$	gamma
HVAC Room (Rm. 504)	104/50	104/50	104	atm.	atm.	80/20	80	100	$1.3x10^4$	gamma
HVAC Room (Rm. 507)	104/50	104/50	104	atm.	atm.	80/20	80	100	$1.3x10^4$	gamma

NA Not applicable

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location	Temperature			Pressure		Relative Humidity		Cumulative Radiation[2] Dosage		
(Environmental Designator)	Normal Range (max/min. °F)	Abnormal (max/min. °F)	Accident (°F)	Normal	Accident	Normal Range (max/min. %)	Accident (%)	Normal (rads)	Accident (rads)	Radiation Type
HVAC Room (Rms. 508, 509)	104/50	104/50	104	atm.	atm.	80/20	80	100	$1.3x10^4$	gamma
Radiation Monitor (Rm. 510)	104/50	104/50	104	atm.	atm.	80/20	80	100	$1.3x10^4$	gamma
4. Fuel Handling Building										
HVAC Supply Subsystems) (Rm. 002)	104/65	117/62	120	slightly negative	slightly negative	80/62	100	10^3	$1.2x10^4$	gamma
HVAC Room (Rm. 003)	120/65	120/62	120	slightly negative	slightly negative	80/20	80	10^3	100	gamma
ECCS Cubicles (Rms. 004, 005, 006)	104/65	104/62	120	slightly negative	slightly negative	80/20	80	10^3	$8.9x10^6$	gamma
Recirculation Valve Rooms (Rms. 007, 008, 009)	104/65	104/62	120	slightly negative	slightly negative	80/20	80	10^3	$9.8x10^6$	gamma
Spray Additive Tank Rooms (Rms. 007A, 008A, 009A)	104/65	120/62	120	slightly negative	slightly negative	80/20	100	10^3	$2.1x10^4$	gamma
HVAC Room (Rm. 010)	104/65	107/62	120	slightly negative	slightly negative	80/20	80	10^3	$8.9x10^6$	gamma
HVAC Carbon Filter Room (Rm. 104)	104/65	120/62	120	slightly negative	slightly negative	80/20	100	10^3	100	gamma

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location (Environmental Designator)	Temperature Normal Range (max/min. °F)	Temperature Abnormal (max/min. °F)	Temperature Accident (°F)	Pressure Normal	Pressure Accident	Relative Humidity Normal Range (max/min. %)	Relative Humidity Accident (%)	Cumulative Radiation Dosage[(2)] Normal (rads)	Cumulative Radiation Dosage[(2)] Accident (rads)	Radiation Type
Spent Fuel Pool Pumps (Rms. 106, 107)	104/65	120/62	120	slightly negative	slightly negative	80/20	80	$2x10^3$	100	gamma
Storage Room (Rm. 108)	104/65	119/62	120	slightly negative	slightly negative	80/20	100	10^3	100	gamma
New Fuel Storage (Rm. 203)	104/65	109/62	120	slightly negative	slightly negative	80/20	100	$6x10^3$	100	gamma
HVAC Room (Rm. 204)	104/65	112/62	120	slightly negative	slightly negative	80/20	100	10^3	100	gamma
Platform (Rm. 205)	104/65	112/62	120	slightly negative	slightly negative	80/20	100	10^3	100	gamma
Spent Fuel Pool Hx Room (Rm. 206)	104/65	119/62	120	slightly negative	slightly negative	80/20	100	$6x10^3$	100	gamma
Spent Fuel Pool Hx Room (Rm. 207)	104/65	124/62	120	slightly negative	slightly negative	80/20	100	$6x10^3$	100	gamma
Cask Handling Area (Rm. 303)	104/65	109/62	120	slightly negative	slightly negative	80/20	100	10^3	$1.8x10^3$	gamma
Post-Accident Sampling Room (Rm. 305)	75/65	90/62	120	slightly negative	slightly negative	80/20	100	$3.5x10^4$+++	$1.3x10^3$	gamma

+++ Post-Accident Sampling Panels are shielded and exposed to a normal environment of 10^3 rads.

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location	Temperature			Pressure		Relative Humidity		Cumulative Radiation[2] Dosage		
(Environmental Designator)	Normal Range (max/min. °F)	Abnormal (max/min. °F)	Accident (°F)	Normal	Accident	Normal Range (max/min. %)	Accident (%)	Normal (rads)	Accident (rads)	Radiation Type
5. Mechanical Auxiliary Building										
ECW Sump (Rm. 017)	104/50	109/44	125	slightly negative	0.3 psig	80/20	100	10^3	130	gamma
Boric Acid Transfer Pump Cubicles (Rms. 018, 018A)	104/50	104/44	135	slightly negative	0.4 psig	80/20	100	$2x10^3$	100	gamma
Boron Recycle Room (Rm. 019)	104/50	133/44	240	slightly negative	0.5 psig	80/20	100	10^7	100	gamma
Recycle Holdup Tank (Rms. 20, 24)	104/50	120/44	120	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
Recycle Heat Valve (Rms. 22, 23)	104/50	145/44	150	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
Seal Water HX (Rm. 26)	104/50	111/44	130	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
Valve Room (Rm. 031)	104/50	115/44	240	slightly negative	1.6 psig	80/20	100	10^7	100	gamma
CVCS Valve Cubicle (Rm. 033)	104/50	104/44	140	slightly negative	0.6 psig	80/20	100	10^7	100	gamma
Letdown Reheat Exchanger Room (Rm. 035)	104/50	128/44	240	slightly negative	1.6 psig	80/20	100	10^7	100	gamma

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location (Environmental Designator)	Temperature Normal Range (max/min. °F)	Temperature Abnormal (max/min. °F)	Temperature Accident (°F)	Pressure Normal	Pressure Accident	Relative Humidity Normal Range (max/min. %)	Relative Humidity Accident (%)	Cumulative Radiation[(2)] Dosage Normal (rads)	Cumulative Radiation[(2)] Dosage Accident (rads)	Radiation Type
Positive Displacement Charging Pump (Rm. 037)	135/50	126/44	155	slightly negative	0.6 psig	80/20	100	10^7	100	gamma
Centrifugal Charging Pump (Rm. 039)	135/50	135/44	155	slightly negative	0.6 psig	80/20	100	10^7	100	gamma
Centrifugal Charging Pump (Rm. 041)	135/50	135/44	155	slightly negative	1.1 psig	80/20	100	10^7	100	gamma
CVCS Valve Cubicles (Rm. 044)	104/50	104/44	140	slightly negative	1.1 psig	80/20	100	10^7	100	gamma
Moderating HX (Rm. 045)	104/50	119/44	140	slightly negative	0.8 psig	80/20	100	10^7	100	gamma
Valve Room (Rm. 046)	104/50	119/44	120	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
Valve Room (Rm. 048)	104/50	119/44	245	slightly negative	2.1 psig	80/20	100	10^7	100	gamma
Pump Room (Rm. 049)	104/50	115/44	245	atm.	2.1 psig	80/20	100	10^7	100	gamma
Letdown HX Room (Rm. 050)	104/50	122/44	240	slightly negative	1.6 psig	80/20	100	10^7	100	gamma
LWPS Waste Evaporator (Rm. 053B)	104/50	130/44	350	atm.	2.0 psig	80/20	100	10^7	100	gamma

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location (Environmental Designator)	Temperature Normal Range (max/min. °F)	Temperature Abnormal (max/min. °F)	Temperature Accident (°F)	Pressure Normal	Pressure Accident	Relative Humidity Normal Range (max/min. %)	Relative Humidity Accident (%)	Cumulative Radiation[2] Dosage Normal (rads)	Cumulative Radiation[2] Dosage Accident (rads)	Radiation Type
High Activity Spent Resin Storage Tank (Rm. 054)	104/50	120/44	125	slightly negative	0.3 psig	80/20	100	$2x10^9$	100	gamma
Reactor Make-up Water Pump Cubicles (Rm. 062)	104/50	104/44	125	slightly negative	1.2 psig	80/20	100	$2x10^3$	$1.2x10^4$	gamma+beta
Refueling Water Storage Tank Room (Rm. 063)	104/50	111/44	130	slightly negative	2.4 psig	80/20	100	$6x10^3$	$1.6x10^4$	gamma+beta
Non Radioactive Pipe Chase (Rm. 064)	104/50	133/44	125	slightly negative	0.3 psig	80/20	100	10^3	$3.5x10^5$	gamma
Electrical Equipment Room (Rm. 065)	104/50	131/44	135	slightly negative	1.1 psig	80/20	100	100	$1.3x10^3$	gamma
Essential Chiller & CCW Pump Room (Rms. 067, 067E, 067F)	115/50	115/44	120	slightly negative	1.1 psig	80/20	100	10^3	130	gamma
Corridor (Rms. 067A, 067B)	104/50	126/44	170	slightly negative	1.1 psig	80/20	100	10^3	100	gamma
Corridor (Rm. 067C)	104/50	125/44	195	slightly negative	1.6 psig	80/20	100	10^3	100	gamma
Corridor (Rm. 067D)	104/50	126/44	170	slightly negative	1.1 psig	80/20	100	10^3	$1.3x10^3$	gamma

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location	Temperature			Pressure		Relative Humidity		Cumulative Radiation[2] Dosage		
(Environmental Designator)	Normal Range (max/min. °F)	Abnormal	Accident (°F)	Normal	Accident	Normal Range (max/min. %)	Accident (%)	Normal (rads)	Accident (rads)	Radiation Type
Valve Room (Rm. 068E)	104/50	106/44	125	slightly negative	0.3 psig	80/20	100	$6x10^3$	100	gamma
Charcoal Absorber Room (Rm. 068F)	85/50	105/44	125	slightly negative	0.3 psig	80/20	100	10^7	$1.3x10^3$	gamma
Guard Bed Cubicle (Rm. 068H)	85/50	95/44	125	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
H_20 Removal Skid Room (Rm. 068K)	104/50	109/44	125	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
Pump Room (Rm. 072)	104/50	114/44	120	slightly negative	0.3 psig	80/20	100	$6x10^3$	100	gamma
Boric Acid Tank Room (Rm. 076)	104/50	114/44	170	slightly negative	0.3 psig	80/20	100	$2x10^3$	100	gamma
Valve Room (Rm. 079)	104/50	105/44	140	slightly negative	0.3 psig	80/20	100	$6x10^3$	100	gamma
Valve Room (Rm. 079A)	104/50	134/44	140	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
Valve Room (Rm. 079B)	104/50	144/44	240	slightly negative	1.6 psig	80/20	100	10^7	100	gamma
Valve Room (Rm. 079C)	104/50	128/44	140	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
Cooling Water Heat Exchanger (Rm. 106)	104/50	113/44	125	slightly negative	0.3 psig	80/20	100	10^3	100	gamma

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location	Temperature			Pressure		Relative Humidity		Cumulative Radiation(2) Dosage		
(Environmental Designator)	Normal Range (max/min. °F)	Abnormal (max/min. °F)	Accident (°F)	Normal	Accident	Normal Range (max/min. %)	Accident (%)	Normal (rads)	Accident (rads)	Radiation Type
Non Radioactive Piping (Rm. 106A)	104/50	119/44	120	slightly negative	0.3 psig	80/20	100	10^3	130	gamma
Non Radioactive Piping (Rm. 107)	104/50	112/44	195	slightly negative	2.4 psig	80/20	100	10^3	100	gamma
Radioactive Pipe Pen. Area (Rm. 108)	104/50	131/44	220	slightly negative	2.4 psig	80/20	100	10^7	$1.3x10^4$	gamma
Corridor (Rm. 108A)	104/50	114/44	220	slightly negative	2.4 psig	80/20	100	10^3	$1.3x10^4$	gamma
Non Radioactive Pipe Chase (Rm. 108B)	104/50	117/44	160	slightly negative	2.4 psig	80/20	100	10^3	100	gamma
Radioactive Pipe Chases (Rm. 108C)	104/50	124/44	230	slightly negative	2.6 psig	80/20	100	10^7	$2.6x10^5$	gamma
Radioactive Pipe Chases (Rm. 108D)	104/50	117/44	175	slightly negative	2.4 psig	80/20	100	10^7	$2.6x10^5$	gamma
Electrical Chase (Rm. 108E)	104/50	111/44	160	slightly negative	2.4 psig	80/20	100	10^3	100	gamma
Drywaste Compactor (Rm. 109)	104/50	111/44	155	slightly negative	0.3 psig	80/20	100	$6x10^3$	100	gamma
Radioactive Pipe Chase (Rms. 110, 110A)	104/50	121/44	170	slightly negative	2.4 psig	80/20	100	10^7	100	gamma

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location (Environmental Designator)	Temperature Normal Range (max/min. °F)	Temperature Abnormal (max/min. °F)	Temperature Accident (°F)	Pressure Normal	Pressure Accident	Relative Humidity Normal Range (max/min. %)	Relative Humidity Accident (%)	Cumulative Radiation[2] Dosage Normal (rads)	Cumulative Radiation[2] Dosage Accident (rads)	Radiation Type
LASR Storage Tank & Pump Room (Rm. 112)	104/50	105/44	120	slightly negative	0.8 psig	80/20	100	$3x10^7$	100	gamma
HVAC Room (Rm. 206B)	75/72	104/50	120	slightly negative	0.3 psig	80/20	100	10^3	130	gamma
Non Radioactive Pipe Penetration (Rm. 216)	104/50	125/44	125	slightly negative	0.3 psig	80/20	100	10^3	$9.7x10^5$	gamma
Radwaste Control Room (Rm. 217)	78/50	113/44	125	slightly negative	0.3 psig	80/20	100	10^3	100	gamma
Stairway (Rm. 217A)	78/50***	104/44	125	slightly negative	0.3 psig	80/20	100	10^3	100	gamma
Electrical Chase Room (Rm. 217B)	104/50	116/44	125	slightly negative	0.3 psig	80/20	100	10^3	100	gamma
Hallway (Rm. 218)	78/50	112/44	125	slightly negative	0.3 psig	80/20	100	10^3	130	gamma
Corridor (Rm. 218C)	104/50	117/44	125	slightly negative	0.3 psig	80/20	100	10^3	100	gamma
Corridor (Rm. 218K)	104/50	120/44	140	slightly negative	0.3 psig	80/20	100	10^3	100	gamma
Bypass Transfer Area (Rm. 219A)	75/72	84/44	120	slightly negative	0.3 psig	80/20	100	100	100	gamma
CVCS Valve Cubicle (Rm. 226)	104/50	104/44	125	slightly negative	0.3 psig	80/20	100	10^7	100	gamma

*** There is no HVAC or equipment in this area; therefore, the temperatures listed are based upon the adjacent room temperatures.

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location	Temperature			Pressure		Relative Humidity		Cumulative Radiation[(2)] Dosage		
(Environmental Designator)	Normal Range (max/min. °F)	Abnormal (max/min. °F)	Accident (°F)	Normal	Accident	Normal Range (max/min. %)	Accident (%)	Normal (rads)	Accident (rads)	Radiation Type
Counting Room (Rm. 231)	79/68	120/44	125	slightly positive	0.3 psig	80/20	100	10^3	100	gamma
Volume Control Tank (Rm. 233)	104/50	116/44	120	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
Rad. Chem. Lab and Sample Room (Rm. 234)	75/72	113/44	125	slightly negative	0.3 psig	80/20	100	$6x10^3$	100	gamma
Sample Room (Rm. 235)	75/72	115/44	125	slightly negative	0.3 psig	80/20	100	$3x10^4$	100	gamma
Boron Analyzer (Rm. 235A)	75/72	97/44	120	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
Valve Area (Rm. 238)	104/50	116/44	120	slightly negative	0.3 psig	80/20	100	$6x10^3$	100	gamma
Valve Room (Rm. 238K)	104/50	116/44	120	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
Valve Area (Rm. 238R)	104/50	116/44	120	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
Filter Area (Rm. 243)	104/50	111/44	120	slightly negative	0.3 psig	80/20	100	$6x10^3$	100	gamma
Filter Area (Rms. 243A, 243L)	104/50	112/44	140	slightly negative	0.3 psig	80/20	100	$2x10^8$	100	gamma

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location (Environmental Designator)	Temperature Normal Range (max/min. °F)	Temperature Abnormal (max/min. °F)	Temperature Accident (°F)	Pressure Normal	Pressure Accident	Relative Humidity Normal Range (max/min. %)	Relative Humidity Accident (%)	Cumulative Radiation(2) Dosage Normal (rads)	Cumulative Radiation(2) Dosage Accident (rads)	Radiation Type
Mixed Bed Demineralizer Cubicles (Rms. 244A-T)	104/50	116/44	125	slightly negative	0.3 psig	80/20	100	$2x10^9$	100	gamma
HVAC Equipment Room (Rm. 324)	104/50	94/44	120	slightly negative	2.4 psig	80/20	100	10^3	100	gamma
Corridor (Rm. 324A)	104/50	121/44	125	slightly negative	2.4 psig	80/20	100	10^3	130	gamma
CCW Surge Tank Room (Rm. 324B)	104/60	114/44	125	slightly negative	2.4 psig	80/20	100	10^3	130	gamma
Decontamination (Rms. 325, 325A)	85/50	107/44	125	slightly negative	0.3 psig	80/20	100	$2x10^3$	130	gamma
Personnel Hatch Area (Rm. 326)	104/50	112/44	125	slightly negative	0.3 psig	80/20	100	10^3	$1.7x10^6$	gamma
Valve Operating Area (Rm. 327)	104/50	118/44	120	slightly negative	0.3 psig	80/20	100	10^7	$1.3x10^3$	gamma
HVAC Room (Rm. 327A)	104/50	104/44	125	slightly negative	1.6 psig	80/20	100	10^3	$1.3x10^3$	gamma
Hydrogen & Radiation Monitors (Rm. 328)*	104/50	104/44	125	slightly negative	0.3 psig	80/20	100	10^3	$1.7x10^6$	gamma

* Hydrogen and radiation monitors are shielded and exposed to an accident environment of $2.3x10^3$ rads.

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location (Environmental Designator)	Temperature Normal Range (max/min. °F)	Temperature Abnormal (max/min. °F)	Temperature Accident (°F)	Pressure Normal	Pressure Accident	Relative Humidity Normal Range (max/min. %)	Relative Humidity Accident (%)	Cumulative Radiation[(2)] Dosage Normal (rads)	Cumulative Radiation[(2)] Dosage Accident (rads)	Radiation Type
Concentrate Storage Tank (Rm. 331)	104/50	116/44	140	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
Corridor (Rm. 331A)	104/50	109/44	140	slightly negative	0.3 psig	80/20	100	$6x10^3$	100	gamma
Radwaste Mixing Tank (Rm. 334)	104/50	111/44	125	slightly negative	0.3 psig	80/20	100	10^7	100	gamma
Volume Control Tank Operating Room (Rm. 335)	104/50	107/44	125	slightly negative	0.3 psig	80/20	100	$6x10^3$	130	gamma
6. Diesel-Generator Building										
Engine Room, operating (Rms. 001, 002, 003)	104/50	120/42	120	slightly positive	slightly positive	100	100	100	100	gamma
Stairwell (Rms. 004, 005, 006)	104/50	120/42	120	atm.	atm.	100/20	100	100	100	gamma
Intake Filter Rooms (Rms. 101, 102, 103)	104/50	120/42	120	slightly positive	slightly positive	100/20	100	100	100	gamma
Diesel Oil Storage Room (Rms. 107, 108, 109)	104/50	113/29	120	slightly negative	slightly negative	100/20	100	100	100	gamma
Stairwell (Rms. 110, 111, 112)	104/50	104/29	120	slightly negative	slightly negative	100/20	100	100	100	gamma

TABLE 3.11-1 (Continued)

ENVIRONMENTAL CONDITION

Location	Temperature			Pressure		Relative Humidity		Cumulative Radiation[2] Dosage		
(Environmental Designator)	Normal Range (max/min. °F)	Abnormal (max/min. °F)	Accident (°F)	Normal	Accident	Normal Range (max/min. %)	Accident (%)	Normal (rads)	Accident (rads)	Radiation Type
Intake Air (Rms. 207, 208, 209)	104/50	104/29	120	slightly negative	slightly negative	100/20	100	100	100	gamma
7. Turbine-Generator Building										
General Areas	110/50	147/43	NA	slightly positive	NA	80/20	NA	100	100	gamma
8. Miscellaneous Buildings										
ECWIS (Areas without an active train are maintained above 50°F)	104/34	104/34	104/34	slightly negative	slightly negative	80/20	80	100	100	gamma

NA Not applicable

TABLE 3.11-3

SAFETY-RELATED EQUIPMENT (INSIDE CONTAINMENT)

OPERATIONAL AND QUALIFICATION REQUIREMENTS

Equipment	Purpose	Qualification Requirements
Class 1E electric cables	Supply power, instrumentation and control of those devices required to function during and after an accident	IEEE 383-1974, RG 1.131, and NUREG-0588, Rev. 1
Electrical penetration assemblies	Provide means of electrical access into Containment without impairing Containment integrity	IEEE 317-1976, RG 1.63, and NUREG-0588, Rev. 1
Safety-related valves[1]	Containment isolation and various other system-dependent safety-related functions	IEEE 382-1972, RG 1.73, and NUREG-0588, Rev. 1
Reactor Containment fan cooler motors	Containment heat removal	IEEE 334-1974, RG 1.40, and NUREG-0588, Rev. 1
Containment hi-range area radiation monitors	Post-Accident radiation Monitoring	IEEE 323-1974, RG 1.97, and NUREG-0588, Rev. 1
Containment water level instrumentation	Post-Accident Monitoring	IEEE 323-1974, RG 1.97, and NUREG-0588, Rev. 1
Reactor vessel water level instrumentation	Post-Accident Monitoring	IEEE 323-1974, RG 1.97, and NUREG-0588, Rev. 1
Post-Accident monitoring transmitters/sensors (RG 1.97, Category 1 and Category 2	Post-Accident Monitoring	IEEE 323-1974, RG 1.97, and NUREG-0588, Rev. 1

[1]This includes only electrically operated valves contained in electrical EQCPs.

TABLE 3.11-4

SAFETY-RELATED EQUIPMENT (OUTSIDE CONTAINMENT HARSH ENVIRONMENT) OPERATIONAL AND QUALIFICATION REQUIREMENTS

Equipment	Purpose	Qualification Requirements
Class 1E electric cables	Supply power, instrumentation, and control to those devices required to function during and after an accident	IEEE 383-1974, IEEE 323-1974 as modified by RG 1.131, and NUREG-0588, Rev. 1
Motors	Drive for pumps, fans, etc.	IEEE 334-1971, RG 1.40, and NUREG-0588, Rev. 1
Safety-related valves[1,2]	Containment isolation and various other system-dependent safety-related functions	IEEE 382-1972, RG 1.73, and NUREG-0588, Rev. 1
Hydrogen monitors	Post-Accident Hydrogen Monitoring	IEEE 323-1974 and NUREG-0588, Rev. 1
Post-Accident Monitoring transmitters/ sensors (RG 1.97, Category 1 and Category 2)[1]	Post-Accident Monitoring	IEEE 323-1974 and NUREG-0588, Rev. 1
Radiation monitors Class 1E and RG 1.97, Category 1 and 2[1]	Post-Accident monitoring and ventilation system control post-DBA	IEEE 323-1974, RG 1.97, and NUREG-0588, Rev. 1
Auxiliary shutdown Station panels	Control of Class 1E equipment for shut-down from outside main control room	IEEE 323-1974, Refer to Section 3.11.2
Safety-related instrumentation[1]	Various system-dependent safety-related functions)	IEEE 323-1974 and NUREG-0588, Rev. 1

[1]Some equipment may be located in a mild environment. (Refer to Section 3.10 and 3.11 for qualification requirements of equipment located in mild environment).

[2]This includes only electrically operated valves contained in electrical EQCPs.

TABLE 3.11-5

SAFETY-RELATED APPLICATIONS
OF POLYETHYLENE CABLE

Polyethylene is used in Rockbestos cables as listed below. All cables listed below are tested and qualified to IEEE 383-1974 as modified by RG 1.89 (11/74).

Cable Type	Insulation Material	Jacket Material
Instrumentation	Cross-linked polyethylene	Hypalon / Chlorosulfonated Polyethylene
Triaxial	Cross-linked polyethylene	Cross-linked polyethylene
Coaxial 48 vdc Annunciator	Cross-linked polyethylene	Cross-linked polyethylene
Multi-Conductor Control and Instrumentation	Cross-linked polyethylene	Hypalon/ Chlorosulfonated Polyethylene
Multi-Conductor Shielded	Cross-linked polyethylene	Hypalon/ Chlorosulfonated Polyethylene
Thermocouple Extension	Cross-linked polyethylene	Hypalon/ Chlorosulfonated Polyethylene

CN-3036

CN-3036

3.11N ENVIRONMENTAL DESIGN OF ELECTRICAL EQUIPMENT (NSSS)

This section presents information to demonstrate that the safety-related electrical equipment of the Engineered Safety Features (ESFs) and the Reactor Protection Systems (RPSs) are capable of performing their designated safety-related functions while exposed to applicable normal, abnormal, test, accident, and post-accident environmental conditions. The information presented includes identification of the safety-related equipment that is within the scope of the Westinghouse Nuclear Steam Supply System (NSSS). For each item of equipment, the applicable environmental parameters and a description of the qualification process employed to demonstrate the required environmental capability are provided. The seismic qualification of NSSS safety-related electrical equipment is presented in Section 3.10N.

3.11.N.1 Equipment Identification and Environmental Conditions

A complete list of safety-related electrical equipment within the NSSS scope of supply that is required to function during and subsequent to an accident was provided in the applicant's 10CFR50.49 submittal. In addition, this submittal provided the equipment qualification environmental parameters for normal, abnormal, and accident conditions and qualified life. This listing is maintained in accordance with plant procedures. A list of all Category 1 and 2 post-accident monitoring equipment (in response to Regulatory Guide [RG] 1.97, Rev.2) that is included in the equipment qualification program is provided in Table 7.5-1. CN-2861

3.11N.2 Qualification Tests and Analysis

3.11.N.2.1 Environmental Qualification Criteria. The methods of meeting the general requirements for environmental design and qualification of safety-related equipment as described by General Design Criteria (GDC) 1, 2, 4, and 23 are described in Section 3.1. Additional specific information concerning the implementation of GDC 23 is provided in Section 7.2. The general methods of implementing the requirements of Appendix B to 10CFR Part 50 are described in the Westinghouse Water Reactor Division Quality Assurance Plant (WCAP-8370). Recommendations contained in RGs 1.40, 1.73, and 1.89 concerning environmental qualification are met.

Westinghouse meets the Institute of Electrical and Electronic Engineers (IEEE) Standard 323-1974 by either type test, analysis, or an appropriate combination of these methods. Westinghouse meets this commitment employing the methodology described in WCAP-8587 (Ref. 3.11N-1). This WCAP was reviewed and accepted by the Nuclear Regulatory Commission (NRC) through the issuance of a Safety Evaluation Report (SER) on November 10, 1983.

Commensurate with the restrictions placed on time margin, plant specific accident conditions, maintenance and surveillance programs, and additional equipment-specific supporting information as delineated by the SER, the NRC had concluded that WCAP-8587 complies with the NRC environmental requirements as codified by 10CFR50.49 and its subordinate RGs, NUREGs and IEEE Standards.

3.11N.2.2 Performance Requirements for Environmental Qualification. In response to the NRC staff request for additional detailed information on the qualification program, Westinghouse submitted Supplement 1 to WCAP-8587. This supplement contains an equipment qualification data package (EQDP) for every item of safety-related electrical equipment supplied by Westinghouse within the NSSS scope of supply. Table 3.10N-1 identifies the equipment supplied by Westinghouse for this application and identifies the applicable EQDP.

Each EQDP contains a section entitled, "Performance Requirements." This specification establishes the safety-related functional requirements of the equipment to be demonstrated under normal, abnormal, test, accident, and post-accident conditions. The environmental qualification parameters (e.g., temperature, humidity, pressure, radiation, etc.) employed by Westinghouse for generic qualification purposes are also identified in the specification, as applicable.

3.11N.2.3 Methods and Procedures for Environmental Qualification. WCAP-8587 describes the methodology employed by Westinghouse for qualification of safety-related electrical equipment. Each EQDP (Supplement 1, WCAP-8587) contains a description of the qualification plan for its associated piece of equipment. Qualification may be demonstrated by either type test, operating experience, analysis, or a combination of these methods.

3.11N.3 Qualification Program Results

Qualification program results are summarized in the various station records. CN-2861

3.11N.4 Loss of Ventilation

Refer to Section 3.11.4.

3.11N.5 Estimated Chemical and Radiation Environment

The radiation and chemical environments for which the NSSS scope equipment is qualified are defined in the performance specification of the applicable EQDP contained in Supplement 1, WCAP-8587. As discussed in Section 3.11.5, the spray pH design conditions have changed due to the spray additive deletion modification. WCAP-12477 discusses the suitability to the new spray pH design conditions.

REFERENCE

Section 3.11N:

3.11N-1 WCAP-8587, Revision 6A, 1983, Butterworth, G. and Miller, R. B., “Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment”.

3.11N-2 WCAP-8587, Supplemental 1, “Equipment Qualification Data Packages”.

3.12 CONFORMANCE WITH NRC REGULATORY GUIDES

The details of South Texas Project Electric Generating Station (STPEGS) conformance to Nuclear Regulatory Commission (NRC) Regulatory Guides (RGs) are discussed in appropriate sections on individual basis.

The STPEGS Preliminary Safety Analysis Report (PSAR) addressed RGs 1.1 through 1.88. The latest revisions of these RGs are addressed in this section in accordance with RG implementation criteria. Similarly, new RGs issued since the PSAR are discussed herein in accordance with implementation criteria.

Table 3.12-1 provides a cross reference matrix of each RG and the sections where it is discussed. In addition, the status of STPEGS compliance is summarized in the table.

3.12.1 Regulatory Guide 1.121

The STPEGS position on RG 1.121, "Bases for Plugging Degraded Steam Generator Tubes," is provided below:

Position C.1

STPEGS interprets the term "unacceptable defects" to apply to those imperfections resulting from service-induced mechanical or chemical degradation of the tube walls which have penetrated to a depth in excess of the plugging limit.

Position C.2.b

In cases where sufficient inspection data exists to establish a degradation allowance, the rate used will be an average time-rate determined from the mean of the test data.

Position C.3.d(1) and C.3.d(3)

The combined effect of these requirements would be to establish a maximum permissible primary-to-secondary leak rate which may be below the threshold of detection with current methods of measurement. STPEGS has determined the maximum acceptable length of a through-wall-crack based on secondary pipe break accident loadings which are typically twice the magnitude of normal operating pressure loads. STPEGS will use a leak rate associated with the crack size determined on the basis of accident loadings.

Position C.3.e(6)

STPEGS will supply computer code names and references rather than the actual codes.

Position C.3.f(1)

STPEGS will establish a minimum acceptable tube wall thickness (plugging limit) based on structural requirements and consideration of loadings, measurement accuracy and, where applicable, a degradation allowance as discussed in this position and in accordance with the general intent of this guide. Analyses to determine the maximum acceptable number of tube failures during a postulated condition are normally done to entirely different bases and criteria and are not within the scope of this guide.

Position C.3.f(4)

Where requirements for minimum wall are markedly different for different areas of the tube bundle (e.g., U-bend area versus straight length in Westinghouse designs) two plugging limits may be established to address the varying requirements in a manner which will not require unnecessary plugging of tubes.

3.12.2 Regulatory Guide 1.2

The STPEGS position on RG 1.2, "Thermal Shock to Reactor Vessels", is given below:

Westinghouse follows all recommendations of the guide. Regulatory Position C.1 is followed by Westinghouse analytical and experimental programs as well as by participation in the Heavy Section Steel Technology (HSST) Program at Oak Ridge National Laboratory.

Analytical techniques have been developed by Westinghouse to perform fracture evaluations of reactor vessels under thermal shock loadings.

Under the HSST Program 6-inch-thick, 39-in. outside diameter (OD) steel pressure vessels containing carefully prepared and sharpened surface cracks are being tested. Test conditions include both hydraulic internal pressure loadings and thermal shock loadings. The objective of this program is to validate analytical fracture mechanics techniques and demonstrate quantitatively the margin of safety inherent in reactor pressure vessels.

A number of vessels have been tested under hydraulic pressure loadings, and results have confirmed the validity of fracture analysis techniques. The results and implications of the hydraulic pressure tests are summarized in Oak Ridge National Laboratory Report No. ORNL-TM-5090.

Four thermal shock experiments have been completed and are now being evaluated. For representative conditions, flaws are shown to initiate and arrest in a predictable manner.

Fracture toughness testing of irradiated compact tension fracture toughness specimens has been completed. The complete post-irradiation data on 0.394-inch, 2-inch, and 4-inch-thick specimens are now available from the HSST program. Both static and dynamic post-irradiation fracture toughness data have been obtained. Evaluation of the data obtained to date on material irradiated to fluences between 2.2 and 4.5 x 10^{19} n/cm^2 indicates that the reference toughness curve as contained in the ASME B&PV Code Section III remains a conservative lower bound for toughness values for pressure vessel steels.

Details of progress and results obtained in the HSST program are available in the HSST Program Quarterly Progress Reports, issued by Oak Ridge National Laboratory.
Regulatory Position C.2 is followed in as much as no significant changes have been made in approved core or reactor designs.

Regulatory Position C.3 is followed since the vessel design does not preclude the use of an engineering solution to assure adequate recovery of the fracture toughness properties of the vessel material. If additional margin is needed, the reactor vessel can be annealed at any point in its service life. This solution is already feasible, in principle, and could be performed with the vessel in place.

TABLE 3.12-1

REGULATORY GUIDE MATRIX

ABBREVIATIONS:

A Conform to guide
B Conform to intent of guide
C Take partial exception to guide
D Alternate approach is used
ER Environmental Report
FC For comment
NA Not Applicable
G No commitment to this guide

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	6.2.2.3.5 6.3.2.2	Rev 0 (11/70)	A See Note 62
1.2	Thermal Shock to Reactor Pressure Vessels	3.12.2	Rev 0 (11/70)	D See Note 82
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors			NA See Note 1
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors			D See Note 100
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors			NA See Note 1
1.6	Independence Between Redundant Standby (onsite) Power Sources and Between their Distribution System	Table 7.1-1 Figure 7.1-1 7.6.1.2 8.3.1.2.2 8.3.2.1.1 8.3.2.2.2	Rev 0 (3/71)	A

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.7	Control of Combustible Gas Concentrations in Containment	6.2.5 Table 7.1-1 Figure 7.1-1	Rev 3 (5/2003)	A
1.8	Personnel Selection and Training	13.1.3.1		See Note 96
1.9	Selection of Diesel Generator Set Capacity for Standby Power Supplies	8.3.1.2.3 8.3.1.1.4.2 8.3.1.1.4.7	Rev 2 (12/79)	C (Exception is discussed in 8.3.1.2.3)
1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures	3.8.1.2.2 3.8.1.6.2.3 3.8.1.6.3 3.8.3.2.2 3.8.3.6.2 3.8.4.2.2	Rev 1 (1/73)	C (Exceptions are discussed in UFSAR References) See Note 45
1.11	Instrument Lines Penetrating Primary Reactor Containment	3.1.2.5.6.1 6.2.4.1 7.3.1.1.2 Table 7.1-1 Figure 7.1-1	Rev 0 (3/71)	A
1.12	Instrumentation for Earthquakes	3.7.4.1 Table 7.1-1	Rev 1 (4/74)	D See Note 55
1.13	Spent Fuel Storage Facility Design Basis	3.1.2.6.3.1 3.8.4.2.2 9.1.1.3 9.1.2.3 9.1.4.3	Rev 1 (12/75) FC	A

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.14	Reactor Coolant Pump Flywheel Integrity	5.4.1.5.2 5.4.1.5.3 5.4.1.5.4	Rev 1 (8/75) FC	C See Note 4
1.15	Testing of Reinforcing Bars for Category I Concrete Structures	3.8.1.2.2 3.8.1.6.2.3 3.8.3.2.2 3.8.4.2.2	Rev 1 (12/72)	A
1.16	Reporting of Operating Information Appendix A Technical Specifications		Rev 4 (8/75) FC	B See Note 68
1.17	Protection of Nuclear Power Plants Against Industrial Sabotage		Rev 0 (6/73)	B
1.18	Structural Acceptance Test for Concrete Primary Reactor Containments	3.8.1.7.1 3.8.1.2.2	Rev 1 (12/72)	B
1.19	Nondestructive Examination of Primary Containment Liner Welds	3.8.1.6.4.4 3.8.1.2.2	Rev 1 (8/72)	A
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	3.9.2.4	Rev 2 (5/76)	A
1.21	Measuring, Evaluating, and Reporting Radio-activity in Solid Wastes and Releases of Radio-active Materials in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants	11.5.1.1 12.3.4 12.5.2.2	Rev 1 (6/74)	A

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.22	Periodic Testing of Protection System Actuation Functions	7.1.2.5 Table 7.1-1 7.2.2.2.3.10 7.2.3 7.3.1.2.2 7.3.1.2.2.5.1 7.3.1.2.2.5.4.4 7.4.2.1 7.4.2.2 7.4.2.3 7.4.2.4 7.4.2.5 7.4.2.6 7.4.2.7 8.3.1.1.4.7 8.3.2.2.7 Figure 7.1-1	Rev 0 (2/72)	A
1.23	Onsite Meteorological Programs	2.3.3 Q372.7 Q372.10	Rev 0 (2/72)	C See Table 2.3-23
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure			NA See Note 20
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel-Handling Accident in the Fuel-Handling and Storage Facility for Boiling and Pressurized Water Reactors			D See Note 100

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.26	Quality Group Classifications and Standards for Water-Steam- and Radioactive-Waste-Containing Components of Nuclear Power Plants	3.2.A.2 3.2.B.2 6.6.1 10.4.8.1.5	Rev 3 (2/76) FC	A
1.27	Ultimate Heat Sink for Nuclear Power Plants	2.3.1.2.8 9.2.1.2.1 9.2.5.1.1.5 9.2.5.2 9.2.5.3	Rev 1 (3/74) Rev 2 (1/76) FC	A C See Note 5
1.28	Quality Assurance Program Requirements (Design and Construction)	6.2.5.2.4 Figure 7.1-1		See Note 96
1.29	Seismic Design Classification	3.2.A.1 3.2.B.1 3.8.4.2.2 6.2.5.2.4 6.5.1.5 7.4.2.1 7.4.2.2 7.4.2.3 7.4.2.4 7.4.2.5 7.4.2.6 7.4.2.7 Table 7.1-1 9.1.1.1 9.1.2.1 10.4.8.1.5 Figure 7.1-1	Rev 2 (2/76) FC Rev 3 (9/78)	A A See Note 38

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	3.11.2 6.5.1.5 Table 7.1-1 8.3.1.2.9 8.3.2.2.6 Figure 7.1-1 7.1.2.10	Rev 0 (8/72)	B See Note 79
1.31	Control of Stainless Steel Welding	3.9.5.1 4.5.2.4 5.2.3.4.6 5.3.1.4	Rev 3 (4/78)	B
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants	Table 7.1-1 Figure 7.1-1 7.6.1.2 8.2.1.3 8.3.1.2.4 8.3.2.1.1 8.3.2.1.4 8.3.2.2.3 Appendix 7B	Rev 2 (2/77)	A
1.33	Quality Assurance Program Requirements (Operations)	3.11.2 13.5.1.1		See Note 96
1.34	Control of Electroslag Weld Properties	4.5.2.4 5.2.3.3.2 5.2.3.4.6 5.3.1.4	Rev 0 (12/72)	A
1.35	In-Service Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures	3.8.1.7.3.1 3.8.1.7.3.1.1 3.8.1.2.2	Rev 3 (Proposed 5/79)	C See Note 85
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	4.5.2.4 5.2.3.2.3	Rev 0 (2/73)	A

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	6.1.1.1 4.5.2.5 5.4.2.1.1 5.2.3.4 10.3.6.3	Rev 0 (3/73)	B See Notes 60, 79
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items of Water-Cooled Nuclear Power Plants	6.2.5.2.4		See Notes 96
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants		Rev 0 (3/73) Rev 2 (9/77)	A A See Note 65
1.40	Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants	3.11.2 3.11N.2.1 Table 3.11-3 Table 3.11-4 Table 7.1-1 Figure 7.1-1	Rev 0 (3/73)	A
1.41	Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments		Rev 0 (3/73)	A
1.42	Interim Licensing Policy on As Low As Practicable for Gaseous Radioactive Releases from Light Water-cooled Nuclear Power Reactors		Withdrawn	
1.43	Control of Stainless Steel Cladding of Low-alloy Steel Components	5.2.3.3.2 5.3.1.4	Rev 0 (5/73)	B
1.44	Control of the Use of Sensitized Stainless Steel	3.9.5.1 4.5.2.4 5.2.3.4 5.3.1.4	Rev 0 (5/73)	C See Note 47 B

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems	5.2.5 11.5.2.3.2 Table 7.1-1	Rev 0 (5/73)	B See Note 99
1.46	Protection Against Pipe Whip Inside Containment	None	Withdrawn	G See Note 63
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	7.1.2.6 Table 7.1-1 7.5.4 8.3.1.2.4 8.3.2.2.7 Figure 7.1-1	Rev 0 (5/73)	A
1.48	Design Limits and Loading Combinations for Seismic Category I Fluid System Components	10.4.8.1.5 Table 3.7-1 Table 3.9-2.5	Rev 0 (5/73)	C See Note 30
1.49	Power Levels of Nuclear Power Plants		Rev 1 (12/73)	C See Note 21
1.50	Control of Preheat Temperature for Welding of Low-alloy Steel	5.2.3.3.2 5.3.1.4 10.3.6.2 Q122.10 Q122.18	Rev 0 (5/73)	C See Note 48
1.51	Inservice Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components		Withdrawn	
1.52	Design, Testing, and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Units of Light-Water-Cooled Nuclear Power Plants	6.4.5.2 6.5.1.1.1 6.5.1.1.2 6.5.1.2.1 6.5.1.2.2 Table 6.5-1 9.4.1.1 9.4.1.4 9.4.2.3 9.4.2.4	Rev 2 (3/78)	C See Note 49

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
		12.3.3.3.2 12.3.3.3.3		
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	7.1.2.7 8.3.1.2.8 8.3.2.2.7 Table 7.1-1 Figure 7.1-1 15.0.8	Rev 0 (6/73)	A See Note 73
1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants	6.1.2.1	Rev 0 (6/73)	B
1.55	Concrete Placement in Category I Structures	3.8.1.2.2 3.8.1.6.1.4 3.8.1.6.1.5 3.8.1.6.2.3 3.8.3.2.2 3.8.4.2.2	Rev 0 (6/73)	A
1.56	Maintenance of Water Purity in Boiling Water Reactors			NA See Note 1
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	3.8.2.3 3.8.2.5 3.8.1.2.2	Rev 0 (6/73)	A
1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel	14.2.2.8		See Note 96
1.59	Design Basis Floods for Nuclear Power Plants	2.4.2.2 3.4 3.8.4.2.2	Rev 0 (8/73)	A
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	3.7.1.1	Rev 1 (12/73)	C See Note 44

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.61	Damping Values for Seismic Design of Nuclear Power Plants	3.7.1.3 Table 3.7-1 3.7.3A.15 3.7.3B.15	Rev 0 (10/73)	C See Note 81
1.62	Manual Initiation of Protective Actions	Table 7.1-1 7.2.1.1.3 7.3.1.2.2.7 8.3.1.2.4 8.3.2.2.7 Figure 7.1-1	Rev 0 (10/73)	A
1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants	3.11.2 7.1.2.8 Table 7.1-1 8.3.1.1.4.6 8.3.1.1.5 Figure 7.1-1 Q430.21N Table 3.11-3	Rev 0 (10/73)	A
1.64	Quality Assurance Requirements for the design of Nuclear Power Plants	3.8.4.2.2 Figure 7.1-1		See Note 96
1.65	Materials and Inspections for Reactor Vessel Closure Studs	5.3.1.7 Q121.16	Rev 0 (10/73)	C See Note 50
1.66	Nondestructive Examination of Tubular Products	4.5.2.3 5.2.3.3.2 5.2.3.4.6 Table 5.2-6	Rev 0 (10/73)	C See Note 51
1.67	Installation of Overpressure Protection Systems	3.9.3.3.2 5.4.11.3 Table 7.1-1	Rev 0 (10/73)	A

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.68	Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors	3.9.2.1 Table 7.1-1 14.2.7 Figure 7.1-1	Rev 2 (8/78)	B
1.68.1	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants			NA See Note 1
1.68.2	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	14.2.7	Rev 1 (7/78)	A
1.68.3	Preoperational Testing of Instrument and Control Air Systems	14.2.12.2	Rev 0 (4/82)	C See Note 76
1.69	Concrete Radiation Shields for Nuclear Power Plants	12.3.2.2.1 3.8.1.2.2 3.8.3.2.2 3.8.4.2.2	Rev 0 (12/73)	A
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants	1.1 3.3.1.1 15.0	Rev 2 (9/75)	A See Note 23
1.71	Welder Qualification for Areas of Limited Accessibility	3.9.5.1 4.5.2.4 5.2.3.3.2 5.2.3.4.6 5.3.1.4 10.3.6.2 Q122.13	Rev 0 (12/73)	B See Note 56
1.72	Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin			NA See Note 8

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	3.11.2 3.11N.2.1 Table 3.11-3 Table 3.11-4 Table 7.1-1 Figure 7.1-1 Appendix 7B	Rev 0 (1/74)	A
1.74	Quality Assurance Terms & Definitions		Withdrawn	See Note 96
1.75	Physical Independence of Electric Systems	Figure 7.1-1 7.1.2.2.1 Table 7.1-1 8.3.1.1.4.4 8.3.1.2.7 8.3.1.3 8.3.1.5 8.3.2.2.4 9.5.1.2.2 7.1.2.1.7 7.2.1.1.3 7.2.2.2.3.6 7.3.1.1.2 7.5.4	Rev 2 (9/78)	C See Note 52
1.76	Design Basis Tornado for Nuclear Power Plants	Ref 3.3-5 3.5.1.4 3.8.1.2.2 3.8.4.2.2	Rev 0 (4/74)	A
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors			D See Note 100
1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	2.2 2.2.3.1.1 2.2.3.1.6 6.4.4.2	Rev 0 (6/74)	C See Notes 53 and 98

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	6.3.4.1	Rev 1 (9/75)	C See Note 41
1.80	Preoperational Testing of Instrument Air Systems		Withdrawn	NA See Note 9
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	8.3.1.2.11 8.3.2.2.7	Rev 1 (1/75)	A
1.82	Sumps for Emergency Core Cooling and Containment Spray Systems	6.2.2.1.2 6.2.2.2.3 6.3.4.1	Proposed Rev 1 (5/83)	A
1.83	Withdrawn			N/A See Note 72
1.84	Code Case Acceptability ASME Section III Design and Fabrication	5.2.1.2 5.4.2.1.1 Q210.07N	Rev 24 (7/86)	A See Note 42
1.85	Code Case Acceptability ASME Section III Materials	5.2.1.2 5.3.1.7 5.4.2.1.1 Q210.07N	Rev 24 (7/86)	A See Note 42
1.86	Termination of Operating Licenses for Nuclear Reactors		Rev 0 (6/74)	A See Note 10
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596)			NA
1.88	Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records			See Note 96

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.89	Qualification of Class 1E Equipment for Nuclear Power Plants	3.10N.1.1 3.11.2 3.11.5.2 3.11N.2.1 Table 7.1-1 Figure 7.1-1	Rev 0 (11/74)	B
1.90	In-Service Inspection of Prestressed Concrete Containment Structures with Grouted Tendons			NA See Note 11
1.91	Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites	2.2.3.1.2 2.2.A	Rev 0 (1/75)	A See Note 3
1.92	Combining Model Responses and Spatial Components in Seismic Response Analysis	3.7.2.7 3.7.3A.1.1	Rev 1 (2/76)	A See Note 3
1.93	Availability of Electric Power Sources	8.3.1.2.1 8.3.2.2.7	Rev 0 (12/74)	B See Note 29
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	3.8.4.2.2 3.8.3.2.2 3.8.1.2.2	Rev 1 (4/76)	C See Notes 3, 54
1.95	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release	Table 7.1-1		NA See Note 12
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants			NA See Note 1
1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident	12.3.4 Table 7.1-1 Figure 7.1-1 7.5.1.2 Table 7.5-1 App. 7A App. 7B	Rev 2 (12/80) Rev 3 (5/83)	B See Note 64 B See Note 3

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
		3.11.2 Table 3.11-3 Table 3.11-4 3.11.1 3.11N.1		
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor			NA See Note 1
1.99	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials	5.3.2.1	Rev 1 (4/77)	D See Note 57
	Radiation embrittlement of Reactor Vessel Materials		Rev 2 (5/88)	B See Note 92
1.100	Seismic Qualification of Electric Equipment for Nuclear Power Plants	Table 7.1-1 Figure 7.1-1 3.10.1 3.10.2 3.10.3 3.10.4 3.10N.1.1	Rev 1 (8/77)	B See Note 3
1.101	Emergency Planning for Nuclear Power Plants		Rev 2 (10/81)	A See Note 37
1.102	Flood Protection for Nuclear Power Plants	3.4 3.8.4.2.2	Rev 1 (9/76)	A
1.103	Post-Tensioned Prestressing Systems for Concrete Reactor Vessels and Containments	3.8.1.2.2 3.8.1.6.5.1	Rev 1 (10/76)	A
1.104	Overhead Crane Handling Systems for Nuclear Power Plants	9.1.4.3.1.6 Table 9.1-3	Rev 0 (2/76) FC	B See Note 34
1.105	Instrument Setpoints	Table 7.1-1 Figure 7.1-1 Appendix 7B	Rev 1 (11/76)	B See Notes 3, 28
1.106	Thermal Overload Protection for Electric Motors on Motor-operated Valves	8.3.1.2.12 8.3.2.2.7	Rev 1 (3/77)	A See Note 14

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures			NA See Note 11
1.108	Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems at Nuclear Power Plants	8.3.1.1.4.7 8.3.1.2.10	Rev 1 (8/77)	C See Note 40
1.109	Calculations of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR50, Appendix I	11.A.1 12.4.2 11.A.4.1	Rev 1 (10/77)	A
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors			NA See Note 16
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	2.3.5.2 11.A ER 11.A.3.2.2	Rev 1 (7/77)	A
1.112	Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors		Rev 0-R (5/77)	A
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	11.A.1 11.A.3.2.1	Rev 1 (4/77)	A
1.114	Guidance on Being Operator of the Controls of a Nuclear Power Plant		Rev 1 (11/76)	B See Note 27
1.115	Protection Against Low-Trajectory Turbine Missiles	3.5.1.3	Rev 1 (7/77)	G
1.116	Quality Assurance Requirements for Installation Inspection, and Testing of Mechanical Equipment and Systems		Rev 0-R (5/77)	B See Note 79
1.117	Tornado Design Classification	3.5.1.4	Rev 0 (6/76) FC	A See Note 3

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.118	Periodic Testing of Electric Power and Protection Systems	Table 7.1-1 7.1.2.11 Appendix 7B	Rev. 2 (6/78)	A See Note 69
1.119	Surveillance Program for New Fuel Assembly Designs		Withdrawn	
1.120	Fire Protection Guidelines for Nuclear Power Plants	13.2.3.6 Table 7.1-1	Rev 1 (11/77) FC	D See Note 17
1.121	Bases for Plugging Degraded PWR Steam-Generator Tubes	3.12.1	Rev 0 (8/76) FC	D See Note 58
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	3.7.2.5	Rev 0 (9/76) FC	C See Notes 3, 78
1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants			See Note 96
1.124	Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports		Rev 0 (11/76) FC	A
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants		Rev 0 (3/77) FC	NA See Note 18
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification		Rev 1 (4/78)	D See Note 25
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants		Rev 1 (3/78)	B (Essential Cooling Pond only), See Note 22
1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants	8.3.2.2.5	Rev 1 (10/78)	C See Note 70
1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants	8.3.2.1.4 8.3.2.2.5	Rev 1 (2/78)	D See Note 26

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.130	Design Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports		Rev 0 (7/77) FC	NA See Note 2
1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants	Table 3.11-3 Table 3.11-4 3.11.2 3.11.2.1 8.3.1.4.4.13	Rev 0 (8/77) FC	C See Notes 3, 24
1.132	Site Investigations for Foundations of Nuclear Power Plants		Rev 0 (9/77) FC	NA See Note 2
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors	4.4.6.4	Rev 1 (5/81)	C See Note 86
1.134	Medical Evaluation of Licensed Personnel For Nuclear Power Plants		Rev 2 (11/84) Proposed	A
1.135	Normal Water Level and Discharge at Nuclear Power Plants		Rev 0 (9/77) FC	B See Note 3
1.136	Material for Concrete Containments		Rev 0 (11/77) FC	NA See Notes 2, 13
1.137	Fuel-Oil Systems for Standby Diesel Generators		Rev 0 (1/78) FC	D See Notes 3, 66
1.138	Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants		Rev 0 (4/78) FC	NA See Note 2
1.139	Guidance for Residual Heat Removal		Rev 0 (5/78) FC	A

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.140	Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants	9.4.1.1 9.4.3.1	Rev 0 (3/78)	C See Notes 61, 80
1.141	Containment Isolation Provisions for Fluid System		Rev 0 (4/78) FC	NA See Note 2
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)		Rev 0 (4/78)	NA See Notes 2, 33
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants		Rev 0 (7/78)	G See Note 71
1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants			See Note 96
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants	2.3 15D.1	Rev 0 (8/79)	A
1.146	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants			See Note 96
1.147	Inservice Inspection Code Case Acceptability ASME Section XI Division 1		Rev 6 (5/88)	A See Note 83
1.148	Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants	Table 3.9-23 Table 3.9-24	Rev 0 (3/81)	B
1.149	Nuclear Power Plant Simulators for Use in Operator Training		Rev 4	A See Note 75

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
1.151	Instrument Sensing Lines		Rev 0 (7/83)	NA See Notes 2, 84
1.153	Criteria for Power, Instrumentation, and Control Portions of Safety Systems		Rev 0 (12/85)	NA See Note 2
1.155	Station Blackout		Rev 0 (6/88)	A
1.157	Best-Estimate Calculations of Emergency Core Cooling System Performance		Rev 0 (5/89)	NA See Note 88
1.163	Performance-Based Containment Leak-Test Program	6.2.6.3	Rev 0 (9/95)	C See Note 94
1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	6.5.2.1, 15.1.5, 15.3.3 15.4.8, 15.6, 15.7	Rev. 0 (7/2000)	A
1.190	Calculational and Dosimetry Methods of Determining Pressure Vessel Neutron Fluence	5.3.1.2	Rev. 0 (3/01)	A (see Note 97)
1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants	2.3 15.D.1	Rev.0 (6/2003)	A
1.196	Control Room Habitability at Light-Water Nuclear Power Reactors		Rev. 0 (05/03)	A (See Note 101)
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors		Rev. 0 (05/03)	A (See Note 101)
5.73	Fatigue Management for Nuclear Power Plant Personnel	13.5.1.3.1	Rev. 0 (03/09)	B (see Note 102)
8.4	Direct Reading and Indirect Reading Pocket Dosimeters		Rev 0 (2/73)	B
8.7	Occupational Radiation Exposure Records Systems		Rev 1 (6/92)	B
8.8	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable	12.1.1.2 12.1.2 12.3.4	Rev 3 (6/78)	B

TABLE 3.12-1 (Continued)

Regulatory Guide MATRIX

No.	Regulatory Guide Title	UFSAR Reference	Revision Status On STPEGS	STPEGS Position
8.9	Acceptable Concepts, Models, Equations and assumption for a Bioassay Program		Rev 1 (7/93)	B
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposure ALARA	12.1.1.3	Rev 1 (9/75)	B
8.13	Instruction Concerning Prenatal Radiation Exposure	12.1.1.3	Rev 2 (12/87)	B
8.14	Personnel Neutron Dosimeters		Rev 1 (8/77)	B
8.20	Applications of Bioassay for I-125 and I-131		Rev 1 (9/79)	B
8.26	Applications of Bioassay for Fission and Activation Products		Rev 0 (9/80)	B
8.27	Radiation Protection Training for Personnel at Light Water Cooled Nuclear Power Plants		Rev 0 (3/81)	B
8.28	Audible Alarm Dosimeters		Rev 0 (8/81)	B
8.29	Instruction Concerning Risks from Occupational Radiation Exposure		Rev 1 (2/96)	B
8.34	Monitoring Criteria and Methods to Calculate Occupational Radiation Doses		Rev 0 (7/92)	B
8.35	Planned Special Exposures		Rev 0 (6/92)	B
8.36	Radiation Dose to the Embryo/Fetus		Rev 0 (7/92)	B
8.38	Control of Access to High and Very High Radiation Areas in Nuclear Power Plants		Rev 0 (6/93)	B

TABLE 3.12-1 (Cont'd)

REGULATORY GUIDE MATRIX

NOTES

1. Guide is applicable only to Boiling Water Reactors (BWRs).

2. Guide is not applicable to STPEGS due to implementation date.

3. STPEGS compliance status is indicated, but the guide is not applicable to STPEGS due to implementation date.

4. Pre-spin inspections are considered to be adequate, and post-spin inspection is not performed. RG 1.14 definition of "Excessive Deformation" is not applicable to Westinghouse design. Specification of cross tolling-ratio is unnecessary. Vacuum melting and degassing process or the electro-slag process are not essential to meet balance of RG 1.14 requirements. Each reactor coolant pump flywheel shall be ultrasonically examined over the volume from the inner bore of the flywheel to the circle of one-half the outer radius once every ten years and shall comply with regulatory positions C.4.b(3), (4), and (5).

5. STPEGS is in compliance with Rev. 2 except for meteorological analysis which is in compliance with Rev. 1.

6. Not used.

7. Not used.

8. Guide is not applicable since STPEGS does not have a spray pond.

9. RG 1.80 has been withdrawn by the NRC. The regulatory position is now considered to be covered by RG l.68.3, "Preoperational Testing of Instrument and Control Air Systems."

10. Following plant retirement, STPEGS will be decommissioned in accordance with applicable laws and regulations.

11. STPEGS Containment does not use grouted tendons.

12. Refer to Section 6.4.4.2 for discussion on chlorine within site boundary.

13. RG 1.136 was issued to incorporate the requirements of RGs 1.10, 1.15, 1.18, and 1.19. STPEGS maintains its commitments to RGs 1.10, 1.15, 1.18, and 1.19 as delineated in Table 3.12-1 and the referenced Updated Final Safety Analysis Report (UFSAR) Sections. Therefore a commitment to this guide is not necessary.

14. Thermal overloads for Class 1E motor-operated-valves (MOVs) are alarmed only and are not used for trip/stop at STPEGS. Non-Class 1E MOVs are treated on a case-by-case basis.

TABLE 3.12-1 (Cont'd)

REGULATORY GUIDE MATRIX

NOTES

15. Not used.

16. Guide is not applicable to STPEGS in accordance with 10CFR50, Appendix I, Section II, Paragraph D.

17. STPEGS fire protection utilizes criteria specified in APCSB 9.5-1 Appendix A and 10CFR50, Appendix R (see Fire Hazards Analysis Report [FHAR] Chapter 4).

18. Guide is not applicable since such physical models have not been used for STPEGS.

19. Not used.

20. The STPEGS Gaseous Waste Processing System (GWPS) does not use gas decay tanks.

21. The STPEGS power level will not exceed the maximum specified by the guide. For Standard Thermal Design Procedure (STDP) Safety Analyses evaluated at full power, an initial reactor power level of 102% Rated Thermal Power (RTP) is used in the analyses in accordance with the guide. For Revised Thermal Design Procedure (RTDP) Safety Analysis evaluated at full power, the initial reactor power is the nominal 100% RTP. An uncertainty of 1.3% on the initial power is factored into the DNB correlation as discussed in WCAP-13441. The RTDP alternate methodology was accepted by the NRC as documented in ST-AE-HL-93831. Accidents which use the STDP and RTDP methodology are identified in Table 15.0-2. Westinghouse has performed a sensitivity analysis as documented in ST-UB-HL-1543 which demonstrated that a power uncertainty of 2% has a negligible impact on DNB results.

22. STPEGS complies with the intent of the guide, but the guide is not applicable due to implementation date.

23. The STPEGS UFSAR is written and organized in compliance with the Standard Format and Content Guide.

24. The only exception taken to RG 1.131 is explained in Section 3.11.2.1.

25. Fuel for STPEGS is being provided by Westinghouse. Westinghouse does not comply with RG 1.126; instead, an alternate approach is used as described in WCAP-8218-P-A.

26. Battery maintenance and testing are performed in accordance with Technical Specifications and the Technical Requirements Manual.

TABLE 3.12-1 (Cont'd)

REGULATORY GUIDE MATRIX

NOTES

27. The STPEGS control room layout has two groups of panels in a horseshoe and a separate area with controls located behind one group of the panels. STPEGS meets the intent of this guide by having sufficient operators on duty in the control room to assure visual contact with reactor controls and instrumentation during routine log rounds.

28. RG l.l05 is not discussed explicitly in the UFSAR, however, instrument spans and setpoints are discussed in Sections 7.1.2.1.9, 7.2.2.2.1 and in the Technical Specifications.

29. Electric power availability is discussed in the Technical Specifications although RG l.93 is not explicitly discussed.

30. Compliance to RG 1.48 is explicitly discussed in Table 3.9-2.5.

31. Not used.

32. Not used.

33. The subject RG endorses ACI-349 with exceptions as noted in the RG. Concrete structures (other than the Reactor Containment Building [RCB]) on STPEGS are designed in accordance with ACI-318. A discussion of the significant differences between the ACI-318 and ACI-349 Codes is provided in response to NRC Question 220.30N.

34. This RG was withdrawn by the Commission on August l6, l979 and replaced by NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." Withdrawal of this guide in no way alters any prior or existing licensing commitments based on its use.

35. Not used.

    Not used.

36. Not used.

37. The STPEGS Emergency Plan follows the guidelines of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plan and Preparedness in Support of Nuclear Power Plants".

TABLE 3.12-1 (Cont'd)

REGULATORY GUIDE MATRIX

NOTES

38. The Operations QA Program for operations conforms to the requirements of Rev. 3.

39. Not used.

40. STPEGS takes exception to RG 1.108 as presented in Section 8.3.1.2.10. In addition, diesel generator test frequencies specified in the Technical Specifications are controlled by the Surveillance Frequency Control Program (Technical Specification 6.8.3.r) and do not comply with RG 1.108.

41. HL&P (historical context) does not plan to conduct in situ emergency sump recirculation testing (see Section 6.3.4.l for details). The remainder of the preoperational testing program on the Emergency Core Cooling System (ECCS) and its components was conducted in accordance with RG l.79.

42. RGs 1.84 and 1.85 are periodically revised to incorporate new code cases and revisions to existing code cases. Station personnel will review the revisions to these RGs and comply with the most current revisions of these RGs, as described below.

For components supplied with the NSSS, the following discussion applies:

a. Long-lead-time components for the STPEGS were ordered prior to the original effective date for RGs l.84 and l.85 of July l, l974. Nevertheless, there are no known examples of code cases except those for Classes 2 and 3 components annulled prior to this date, being applied to components except those listed as acceptable by either RG l.84 or l.85, in one of the versions, with the following exception or special consideration:

    1) Code Case l739: This code case was used in the construction of casings for the reactor coolant pumps. Authorization for its use was obtained from the NRC.

    2) Code Case l528: Fracture toughness information for this code case, used in the construction of steam generators and pressurizers, was provided to the NRC (see reference Section of Chapter 5, Section 5.2, and Reference 5.2-7).

b. Westinghouse controls its suppliers to:

   1) Use only the code cases listed in Regulatory Position C.l of the RGs l.84 and l.85 revisions in effect at the time the equipment is ordered, except as allowed in item c. below.

   2) Identify and request permission for use of any code cases not listed in Regulatory Position C.l of the RGs l.84 and l.85 revisions in effect at the time the equipment is ordered, where use of such code cases is needed by the supplier.

   3) Permit continued use of a code case considered acceptable at the time of equipment order, where such code case was subsequently annulled or amended.

c. Westinghouse seeks NRC permission for the use of Class l code cases needed by suppliers and not yet endorsed in Regulatory Position C.l of the RG l.84 and l.85 revisions in effect at the time the equipment is ordered and permits supplier use if NRC permission is obtained or is otherwise assured; e.g., a later version of the RG includes endorsement.

For components not supplied with the NSSS, the requirements of RGs l.84 and 1.85 are met with the following clarifications:

a. Components ordered to a specific version of a code case need not be changed because a subsequent revision to the code case is listed as the approved version in the current revision of the RG.

b. Components ordered to a code case that was previously approved for use need not be changed because the code case is listed as annulled in the current revision of the RG.

43. Not used.

44. Refer to Section 3.7.1.2 for the exceptions to RG 1.60.

45. Refer to Section 3.8.l.6.3 for the exceptions to RG l.l0.

46. Not used.

47. STPEGS conforms to the intent of RG l.44 as noted in Section 4.5.2.4 and takes exception to position C.4(a) as noted in Section 5.2.3.4.

TABLE 3.12-1 (Cont'd)

REGULATORY GUIDE MATRIX

NOTES

48. Refer to Section 5.2.3.3.2, 10.3.6.2, Q122.10, and Q122.18 for the exceptions to RG l.50, specifically positions l(b) and 2.

49.

49. Refer to Table 6.5-l for STPEGS conformance to RG l.52.

50. Westinghouse design is in accordance with RG l.65, except for material and tensile strength guidelines, as noted in Section 5.3.l.7.

5l. Refer to Section 5.2.3.3.2 for the exceptions to RG l.66 positions.

52. Implementation of RG 1.75 is as noted in Section 8.3.1.2.7. RG 1.75 references RG 1.120 as a source for additional criteria for protection against the effects of fires.

53. STPEGS conforms to RG l.78 with the exception that seismically qualified instrumentation for chemicals was not available from the industry at the time of design finalization. Therefore STPEGS is not in full compliance with position C.l2.

54. The QA program for construction conformed to the requirements of RG 1.94, Rev. 1, with the following clarification (this is also described in the STPEGS Quality Assurance Program Description).

    1. The testing frequency of sleeves with filler metal (Cadwelds) complied with UFSAR Sections 3.8.1.6.3 and 3.8.3.6.3.

    2. ANSI N.45.2.5-1974, Section 4.8, states "Pumped concrete must be sampled from the pump line discharge". In lieu of this statement, in-process strength samples of pumped concrete were taken at the delivery point. Correlation tests of air content, slump, and temperature were performed to verify these plastic properties of the concrete at the placement point in accordance with the following frequency requirements:

        a. A minimum of two correlation tests were performed for each pumped placement exceeding 200 yd^3.

        b. Otherwise, a minimum of two correlation tests per week were performed when any individual pumped placement during a week requires delivery of more than one truckload of concrete.

TABLE 3.12-1 (Cont'd)

REGULATORY GUIDE MATRIX

NOTES

c. During a week when a pumped placement exceeding 200 yd^3 was made, the correlation tests performed on that placement satisfied the weekly requirement for performing two correlation tests as specified in Item B, above.

   If the correlation test result showed a concrete property not meeting the specification limits and/or tolerances at the point of placement, the frequency of correlation testing were increased to 100 cubic yards. If two consecutive correlation tests exceeded the specified limit for slump, air content, or temperature, the Constructor documented the condition, notified Bechtel Site Engineering within 24 hours of completion of the placement and returned to control of the concrete by in-process testing at the point of placement per ANSI N45.25-1974.

   "Correlation Tests", "Delivery Point", and "Placement Point" were as defined in ANSI N.45.2.5-1978, Section 1.4.

   Samples and frequency for Cadweld testing was in accordance with ACI-359/ASME Section III, Division 2, issued for trial use and comment in 1973, including addenda 1 through 6, (see Sections 3.8.1.6.3 and 3.8.3.6.3).

If a work activity and contract was for a two-month period or less, an audit was not necessary when a facility preaward audit had been conducted.

The QA program for operations conforms to the requirements of RG 1.94, Rev. 1, with the same clarifications.

55. Refer to Sections 3.7.4.l and 3.7.4.2 for the discussion on seismic instrumentation.

56. Refer to Section 5.2.3.3.2 for Westinghouse alternate approach to RG l.7l. Also, refer to Section l0.3.6.2, for the balance of plant (BOP) conformance to RG l.7l.

57. STPEGS alternate approach to RG l.99 is discussed in Section 5.3.2.l.

58. STPEGS alternate approach to RG l.l2l is discussed in Sections 3.l2.l and 5.4.2.2.

59. Not used.

TABLE 3.12-1 (Cont'd)

REGULATORY GUIDE MATRIX

NOTES

60. With respect to Section 3.l.2 of ANSI N45.2.3-l973, the lighting level of l00 footcandles is interpreted to be guidance. It is the Station's normal practice that the lighting level for determining "metal clean" of accessible surfaces of piping and components is determined by the inspector. Typically, he uses a standard two-cell flashlight supplemented by other lighting as he deems necessary.

6l. See the response to NRC Question 321.4 for the compliance with this RG.

62. RG 1.1 as clarified by NUREG-75/087.

63. RG l.46 has been withdrawn following the issuance of NRC Branch Technical Position (BTP) MEB 3-l and NRC BTP ASB 3-l. Tables 3.6.l-2 and 3.6.l-3 provide a summary of the compliance with MEB 3-l and ASB 3-l.

64. The discussion of STPEGS conformance to RG 1.97 Rev. 2 is presented in Table 7.5-1, Appendix 7A and Appendix 7B. As explained in Appendix 7B, implementation of RG 1.97 requirements was integrated with the Control Room Design Review and was performed using the Westinghouse Owner's Group Emergency Response Guidelines, and conforms with the intent of the RG. CN-3123

65. The QA program during operations conforms to the requirements of Rev. 2.

66. The quality of diesel generator (DG) fuel oil is be checked as identified in Section 9.5.4.4.

67. Not used.

68. Revision 4 of RG 1.16 does not reflect current regulations. STPEGS conforms to regulatory requirements that supercede the requirements of this RG; i.e., 10CFR50.72 and 10CFR50.73.

69. STPEGS conforms to RG 1.118 concerning IEEE 338-1977, Section 6, "Testing Program", however, during ongoing procedures development, additional exceptions and clarifications may be identified. Refer to Section 7.1.2.11 for discussion of conformance to RG 1.118 and IEEE 338-1977.

70. As stated in Section 8.3.2.2.5 the Class 1E DC system at STPEGS is in compliance with RG 1.128 with partial exception.

TABLE 3.12-1 (Cont'd)

REGULATORY GUIDE MATRIX

NOTES

71. STPEGS is not committed to RG 1.143. Most components of the radwaste system are housed within Category I Structures. Those components which are not housed within Category I Structures are either housed in or supplied on structures conforming to NRC BTP ETSB 11-1, Rev. 1, or their failure would not cause radiation release in excess of those considered in the accident analysis. See Sections 11.2, 11.3, and 11.4.

72. RG 1.83 has been withdrawn by the NRC. The inspection program is included in the Steam Generator Program as implemented by Technical Specification Amendments 209 for Unit 1 and 196 for Unit 2. CN-3172

73. STPEGS conforms to RG 1.53 with clarifications as discussed in Section 7.1.2.7.

74. Not used.

75. The STPEGS simulator was operational by February 1986. This simulator will be certified in accordance with RG 1.149 Rev. 4 and ANSI/ANS 3.5-2009.

76. The instrument air system is classified as a non-nuclear safety system. Using the graded approach permitted by RG 1.68, Rev. 2, the instrument air system was acceptance tested using RG 1.68.3 for guidance as identified in Section 14.2.12.2. Based upon the safety evaluation and the unlimited possible number of operating conditions and possible failure modes it is not practical to attempt nor would any significant benefit be derived from attempting to simulate all the possible combinations of line breaks or freezing.

    Testing of the failure mode of each safety-related, air-operated valve was performed in the system preoperational test.

77. Not used.

78. Clarification of the STPEGS position on RG 1.122 is discussed in Section 3.7.2.5.

79. For clarification of applicability to STPEGS during the construction phase, see the Quality Assurance Program Description (QAPD).

80. ANSI N509-1980 (for field testing of atmosphere cleanup units) and ANSI N510-1980 (for in-place testing) are used in conjunction with RG 1.140 in lieu of ANSI N509-1976 and ANSI N510-1975, respectively.
    There are two locations in each unit where HEPA filters were installed in non-nuclear applications: RCB Supplemental Purge Exhaust and Radioactive Vent Header. These installations were to limit migration of particulate to the unit vent and are not required for any

TABLE 3.12-1 (Cont'd)

REGULATORY GUIDE MATRIX

NOTES

safety related function; nor was the application required to credit dose calculations. RG 1.140 and ANSI N509/N510 do not apply.

81. Where RG 1.61 damping valves are used, the design conforms to the RG. However, as stated in Section 3.7.3A.15 there are cases where Pressure Vessel Research Committee (PVRC) damping values are used in lieu of RG 1.61.

82. Refer to Section 5.3.3.6 for discussion on Thermal Shock.

83. RG 1.147 is periodically revised to incorporate new code cases and revisions to existing code cases. Station personnel review the revisions to this RG and complies with the most current revision of this RG with the following clarifications:

    Use of a specific version of a code case need not be changed because a subsequent revision to the code case is listed as the approved version in the current revision of the RG.

    Use of a code case that was previously approved for use need not be changed because the code case is listed as annulled in the current version of the RG.

84. STPEGS is not committed to RG 1.151 since the construction permit date precedes the date of application of the Regulatory Guide.

    The boundary of jurisdiction of the ASME Code, Section III process piping extends to and includes the root valve and the weld from the root valve to the tubing adaptor. The appropriate safety class extends from the root valve to the sensing instrument. Seismic Category I supports are employed for Safety Class 2 and 3 instrument tubing.

    Safety Class 2 and 3 tubing is subject to the following requirements:

    1. Items (i.e., tubing, fittings, and valves) of the safety-related instrument sensing lines are designed for Seismic Category I requirements and are subject to the Project Quality Assurance Program in accordance with 10CFR50, Appendix B. These components are not under the ASME Code, Section III jurisdiction, but they are designed, fabricated, and installed utilizing the ASME Code, Section III as guidance. The safety-related instrument sensing lines are identified as Safety Class 2 or Safety Class 3.

    2. All documentation and material identification as required by Code is maintained. Quality control on weld rod traceability is maintained through welding process data checklists.

TABLE 3.12-1 (Cont'd)

REGULATORY GUIDE MATRIX

NOTES

3. Since the instrument sensing line components are identified as nuclear safety-related in lieu of being ASME Code Section III, these components are not under jurisdiction of the ASME Code, Section III nor under the eventual jurisdiction of the ASME Code Section XI.

85. RG 1.35 Rev. 3 is a proposed Reg. Guide. It governed inservice inspection of tendons through the tenth year surveillances. The fifteenth and twentieth year surveillances complied with the 1992 Edition 1992 Addenda of ASME Section XI, Subsection IWL, as modified and supplemented by 10CFR 50.55a(b)(2)(viii). The twenty-fifth and thirtieth year surveillances will comply with the 2004 Edition No Addenda of ASME Section XI, Subsection IWL, as modified and supplemented by 10CFR50.55a(b)(2)(viii).

86. Section 4.4.6.4 will control and maintain the testing and operability requirements of RG 1.133 in lieu of the Technical Specifications in accordance with letter ST-HL-AE-1923. STP does not commit to the RG 1.133 requirements for Special Reports

87. Not Used.

88. Not applicable due to implementation date. If a new submittal is made which uses the provisions of 10CFR50.46 that allow the use of realistic models as an alternative to the features of Appendix K of 10CFR50, a review will be performed to determine STPEGS compliance with the RG.

89. Not used.

90. Not used.

91. Not used.

92. NRC Generic Letter (GNL 88-011) "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations" required licensees to submit the results of a technical analysis utilizing the methods detailed in Revision 2 of RG 1.99. The results of this analysis show that the STP plant specific P-T curves currently in use (derived from the methodology of RG 1.99 Rev. 1) are more conservative than the P-T limits of RG 1.99, Rev. 2. STP Reactor Vessel Surveillance Capsules, and Pressurized Thermal Shock Analyses are performed in accordance with the methodology of RG 1.99, Rev. 2.

93. Not used.

94. STPEGS complies with RG 1.163 except as modified by approved exemptions.

TABLE 3.12-1 (Cont'd)

REGULATORY GUIDE MATRIX

NOTES

95. Not Used

96. STP's level of commitment and current status pertaining to these Regulatory Guides has been transferred to the Operations Quality Assurance Plan. Any future revisions concerning level of commitment, etc., will be processed as a change to Operations Quality Assurance Plan as per 10CFR50.54(a).

97. Before approval as Regulatory Guide 1.190, this Regulatory Guide was previously referred to as Draft Guide DG-1053. This draft guide may appear in some UFSAR referenced material.

98. STP takes exception to Regulatory Position 13 of NRC Regulatory Guide 1.78, Revision 0 (6/74). Specifically, Position 13 states that "Practice drills should be conducted to ensure that personnel can don breathing apparatus within two minutes." Site-specific analyses demonstrate that operators have at least six minutes before they needed to don breathing apparatus upon nasal detection of a hazardous chemical accident. Therefore, STP will conduct practice drills to assess a sample population to ensure that personnel can don breathing apparatus within "six" minutes.

99. The Containment Atmosphere Radiation Monitor Gaseous Channel was removed from the Reactor Coolant Pressure Boundary Leakage Detection System (Technical Specification 3.4.6.1 and 4.4.6.2.1) by License Amendments 174/162 (Letter ST-AE-NOC-05001425, dated 10/17/2005).

100. For application of the Alternative Source Term (AST), NRC Regulatory Guide 1.183 has superseded Regulatory Guides 1.4, 1.25, and 1.77.

101. STP conforms to RG 1.196 and 1.197 as required by Technical Specification 6.8.3.q the “Control Room Envelope Habitability Program.”

102. STP requested an exemption from certain requirements of the Fitness for Duty Rule for Managing Fatigue during declarations of severe weather conditions (i.e., tropical storm or hurricane force winds). See NOC-AE-09002477 dated October 14, 2009. Until the exemption is approved by the NRC, EGM-09-008, "Enforcement Guidance Memorandum - Dispositioning Violations of NRC Requirements for Work Hour Controls Before and Immediately After a Hurricane Emergency Declaration," dated September 24, 2009 will be followed during severe weather conditions involving tropical storm or hurricane force winds.

TABLE 3.12-1 (Cont'd)

REGULATORY GUIDE MATRIX

NOTES

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