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Volume 2

COMBUSTION **ENGINEERING**

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**3.0   DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND
SYSTEMS****3.1   CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA**

In this section, brief discussions are presented in response to the current General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR 50. These discussions summarize the manner in which the principal design features meet the individual criteria and include references to sections of the safety analysis report where more detailed information is given.

3.1.1   CRITERION 1 - QUALITY STANDARDS AND RECORDS

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

RESPONSE:

The structures, systems, and components described in CESSAR are classified according to their importance in the prevention and mitigation of accidents using the classification system described in ANSI/ANS 51.1. Each safety-related component is given a safety class designation. The codes, standards, and quality control applicable to each component and its safety class designation are identified in Section 3.2. Where applicable, design and fabrication are in accordance with the codes required in 10 CFR 50.55a. The quality assurance program conforms with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants," and is presented in Chapter 17. Chapter 14 describes initial tests and operations to assure performance of installed equipment commensurate with the importance of the safety function.

The design, fabrication, and quality programs for components not included in the ANSI classification system are governed by

industry codes appropriate to the application. Details of conformance to these codes are found in the appropriate CESSAR sections.

**3.1.2 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, tsunamis, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated; (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and, (3) the importance of the safety functions to be performed.

RESPONSE:

The structures, systems, and components designated Seismic Category I are designed to withstand, without loss of function, the effects of any one of the most severe natural phenomena, including flooding, hurricanes, tornadoes, and the Safe Shutdown Earthquake (SSE) (refer to Chapter 2). Design criteria for wind and tornado, flood and earthquake are discussed in Sections 3.3, 3.4, and 3.7, respectively.

The seismic design of safety-related structures, systems, and components is consistent with conservative structural envelopes. These "envelopes" have been selected based on the design basis earthquakes at the majority of potential plant sites in the continental U.S., using current containment structure designs. In the design stage, normal operating and accident loads are appropriately combined with the seismic loads and allowable stress limits and deformations are defined so that safety functions are not jeopardized.

3.1.3 CRITERION 3 - FIRE PROTECTION

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations

such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

RESPONSE:

All pressure boundary components and structures and the attendant auxiliary systems in System 80+ design scope are designed to minimize the probability and effects of fires and explosions. High grade noncombustible and fire resistant materials are used for components located in the containment, components of engineered safety feature systems, and throughout the unit wherever practical. A detailed functional description of the Fire Protection System is provided in Section 9.5.1. | D

3.1.4 CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. | D

RESPONSE:

C-E supplied structures, systems, and components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant-accidents (see Section 3.11).

Where appropriate, Standardized Functional Descriptions will include design requirements to ensure that these structures, systems, and components will be appropriately protected against dynamic effects (including the effects of missiles, pipe | D

whipping, and discharge of fluids) that may result from equipment failures, postulated accidents, and from events and conditions outside the nuclear power unit.

The reactor building is capable of withstanding the effects of missiles originating outside the containment such that no credible missile can result in a LOCA. The control room is designed to withstand such missiles as may be directed toward it and still maintain the capability of controlling the plant. D

3.1.5 CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

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

RESPONSE:

The System 80+ design is based on non-shared systems.

3.1.6 CRITERION 10 - REACTOR DESIGN

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

RESPONSE:

Specified Acceptable Fuel Design Limits (SAFDLs) are stated in Section 4.4.1. Operation within the operating limits (Limiting Conditions for Operation) specified by the Technical Specifications will keep the reactor fuel within the SAFDLs for normal operation and during any Anticipated Operational Occurrence. D

The plant is designed such that operation within Limiting Conditions for Operation with safety system settings not less conservative than the Limiting Safety System Settings prescribed in the Technical Specifications results in confidence that SAFDLs will not be exceeded as a result of any Anticipated Operational Occurrence. Operator action, aided by the control systems and monitored by plant instrumentation, maintains the plant within Limiting Conditions for Operation during normal operation. D

See the following sections for additional information:

- A. Fuel System Design, Section 4.2
- B. Reactor Coolant System, Chapter 5
- C. Shutdown Cooling System, Section 5.4.7
- D. Reactor Protective System, Section 7.2
- E. Analysis of Anticipated Operational Occurrences, Chapter 15
- F. Technical Specifications, Chapter 16

3.1.7 CRITERION 11 - REACTOR INHERENT PROTECTION

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

RESPONSE:

In the power operating range, the combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient, and the moderator pressure coefficient to an increase in reactor power will be a decrease in reactivity; i.e., the inherent nuclear feedback characteristics will not be positive.

The reactivity coefficients for this reactor are discussed in detail in Section 4.3.

3.1.8 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.

RESPONSE:

Power level oscillations do not occur. The effect of the negative power coefficient of reactivity (see GDC 11, Section 3.1.7), together with the coolant temperature program maintained by control of regulating rods and soluble boron, provides fundamental mode stability. Power level is continuously monitored by neutron flux detectors (Chapter 7).

Power distribution oscillations are detected by neutron flux detectors. Axial mode oscillations are suppressed by means of part-strength or full-strength neutron absorber rods. All other modes of oscillation are expected to be convergent. Monitoring and protective requirements imposed by Criteria 10 and 20 are discussed in Sections 3.1.6, 3.1.16 and in Chapter 4. | D

3.1.9 CRITERION 13 - INSTRUMENTATION AND CONTROL

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

RESPONSE:

Instrumentation is provided to monitor significant process variables which can affect the fission process, the integrity of the reactor core, the Reactor Coolant Pressure Boundary (RCPB) and their associated systems. Controls are provided for the purpose of maintaining these variables within the limits prescribed for safe operation. Instrumentation for the containment and its associated systems can be found in the appropriate CESSAR chapters and in the site-specific SAR. The principal process variables to be monitored and controlled are: | D

- A. Neutron flux level (reactor power)
- B. CEA positions
- C. Neutron flux distribution (at various axial positions)
- D. Reactor coolant temperature and pressure
- E. Reactor coolant pump speed
- F. Pressurizer level
- G. Steam generator level and pressure

In addition, Departure from Nucleate Boiling Ratio (DNBR) margin and Local Power Density (LPD) margin, in kW/ft, are also monitored.

The Plant Protection System (PPS) consists of the Reactor Protective System (RPS) and the Engineered Safety Features Actuation System (ESFAS). The RPS is designed to monitor NSSS operating conditions and to initiate reliable and rapid reactor shutdown if monitored variables or combinations of monitored variables deviate from the permissible operating range to a degree that a safety limit may be reached.

The ESFAS is designed to monitor plant variables and to actuate Engineered Safety Feature (ESF) systems during a design basis event.

The following are provided to monitor and maintain control over the fission process during transient and steady state periods over the lifetime of the core:

- A. Redundant channels of ex-core nuclear instrumentation, which constitute the primary means of monitoring the fission process for protection, control and low power operation.
- B. Redundant and diverse CEA position indicating systems for each CEA.
- C. Manual and automatic control of reactor power by means of CEAs.
- D. Manual regulation of coolant boron concentration.
- E. A Boronometer, which determines the boron concentration in the reactor coolant by neutron absorption, provided as a backup to the primary method of determining soluble poison concentration by routine sampling and analysis of reactor coolant.
- F. In-core instrumentation, provided to supplement information on core power distribution and to enable calibration of ex-core flux detectors.

The non-nuclear instrumentation measures temperatures, pressures, flows and levels in the Reactor Coolant System and main steam and auxiliary systems and is used to maintain these variables within the prescribed limits. The instrumentation and control systems are described in detail in Chapter 7. The Boronometer is discussed in Sections 7.7.1.1.7 and 9.3.2 while the process radiation monitor is discussed in Section 9.3.2.

When it is required that a variable be monitored during and after a Design Basis Event (DBE), in addition to normal operation, the

results of analysis of the course of the event are used to ensure that the instruments provided will cover the range anticipated for the event conditions.

3.1.10 CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

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

RESPONSE:

The reactor coolant pressure boundary is defined in accordance with 10 CFR 50.2(v) and ANSI/ANS 51.1 (see response to GDC 55, Section 3.1.48). | D

Reactor Coolant System components are designed to meet the requirements of the ASME Code, Section III. To establish operating pressure and temperature limitations during startup and shutdown of the Reactor Coolant System, the fracture toughness rules defined in the ASME Code, Section III, are followed. Quality control, inspection, and testing are performed as required by ASME Section III and allowable reactor pressure-temperature operations are specified to ensure the integrity of the Reactor Coolant System.

The reactor coolant pressure boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of unit operation including all anticipated transients, and maintain the stresses within applicable limits.

Piping and equipment pressure parts of the reactor coolant pressure boundary are assembled and erected by welding unless applicable codes permit flanged or screwed joints. Welding procedures are employed which produce welds of complete fusion and free of unacceptable defects. All welding procedures, welders, and welding machine operators are qualified in accordance with the requirements of Section IX of the ASME Boiler and Pressure Vessel Code for the materials to be welded. Qualification records, including the results of the procedure and performance qualification tests and identification symbols assigned to each welder are maintained.

The pressure boundary has provisions for in-service inspection in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, to ensure continuance of the structural and leak-tight integrity of the boundary (see response to GDC 32, Section 3.1.28). For the reactor vessel, a material surveillance program conforming with the requirements of Appendix H to 10 CFR 50 is provided.

3.1.11 CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

The Reactor Coolant System (RCS) and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operational occurrences.

RESPONSE:

The design criteria and bases for the reactor coolant pressure boundary are described in the response to Criterion 14.

The operating conditions for normal steady state and transient plant operations are established conservatively. Normal operating limits are selected so that an adequate margin exists between them and the design limits. The plant control systems are designed to ensure that plant variables are maintained well within the established operating limits. The plant transient response characteristics and pressure and temperature distributions during normal operations are considered in the design as well as the accuracy and response of the instruments and controls. These design techniques ensure that a satisfactory margin is maintained between the plant's normal operating conditions, including design transients, and the design limits for the reactor coolant pressure boundary.

Plant control systems function to minimize the deviations from normal operating limits in the event of most Anticipated Operational Occurrences. Where control systems response would be inadequate or fail upon demand, the Plant Protection System functions to mitigate the consequences of such events.

The Plant Protection System functions to mitigate the consequences in the event of accidents. Analyses show that the design limits for the reactor coolant pressure boundary are not exceeded in the event of any ANSI/ANS 51.1 conditions. | D

3.1.12 CRITERION 16 - CONTAINMENT DESIGN

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

RESPONSE

The containment system is designed to protect the public from the consequences of a LOCA, based on the equivalent energy release of a postulated break of reactor coolant piping up to and including a double-ended break of the largest reactor coolant pipe.

The containment vessel, shield building, and the associated Engineered Safety Feature systems are designed to safely withstand all internal and external environmental conditions that may reasonably be expected to occur during the life of the plant, including both short- and long-term effects following a LOCA.

Leak-tightness of the containment system and short- and long-term performance following a LOCA are analyzed in Section 6.2.

3.1.13   CRITERION 17 - ELECTRICAL POWER SYSTEMS

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

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

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

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

RESPONSE:

The System 80+ Standard Design is provided with an onsite electric power system and an offsite electric power system to permit functioning of structures, systems and components important to safety in full compliance with the requirements of this criterion as described in Chapter 8.

The onsite electric power system consists of separate, redundant and independent distribution systems and dedicated power supplies with sufficient capacity, capability, and testability to perform their safety functions assuming a single failure.

The offsite electric power system consists of two physically independent circuits from the station switchyard. Each circuit is immediately available and has sufficient capacity and capability to perform its safety function.

Provisions are made 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.

**3.1.14   CRITERION 18 - INSPECTION AND TESTING OF ELECTRICAL
POWER SYSTEMS**

Electrical power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the system 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 operation sequence that brings the system into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system and the onsite power system.

RESPONSE:

Electrical power systems important to safety are designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and to detect deterioration, if any, of their components. Capability is provided to periodically test the operability and functional performance of the system components. The diesel generator sets will be started and loaded periodically on a routine basis, and relays, switches, and buses will be inspected and tested for operation and availability on an individual basis.

Transfer from normal to emergency sources of power will be made to check the operability of the systems and the full operational sequence that brings the systems into operation.

Refer to Section 8.3.1, 8.3.2 and 16.4.8 for more detailed information.

3.1.15 CRITERION 19 - CONTROL ROOM

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

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

RESPONSE:

All control stations, switches, controllers, and indicators necessary to operate or shut the unit down and maintain safe control of the facility are located in the control room.

The design of the control room permits safe occupancy during abnormal conditions. The employment of non-combustible and fire retardant materials in the construction of the control room, the limitation of combustible supplies, the location of fire fighting equipment, and the continuous presence of a highly trained

operator will minimize the possibility that the control room will become uninhabitable. Shielding is designed to maintain tolerable radiation exposure levels following design basis accidents. The control room will be isolated from the outside atmosphere during the initial period following the occurrence of an accident. The Control Room Ventilation System is designed to recirculate cool control room air as discussed in Sections 9.4.1 and 12.2. Radiation detectors and alarms are provided. Emergency lighting is provided as discussed in Section 9.5.3.

D

Alternate local controls and 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 of the control room through the use of suitable procedures. Refer to Section 7.4.1.1.10. A discussion of the unit's control room is provided in Section 7.7.1.3 with human factors issues discussed in Chapter 18.

A discussion of the hot and cold shutdown capability is provided in Section 7.4 for the systems required for safe shutdown. Discussion regarding adequate radiation protection for the unit's control facilities is provided in Section 6.4 and in Chapter 12.

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3.1.16 CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

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

RESPONSE:

A Plant Protection System (PPS), consisting of a Reactor Protective System (RPS) and an Engineered Safety Features Actuation System (ESFAS), is provided. The RPS automatically initiates a reactor trip when any of the monitored process variables reach a trip setpoint. The ESFAS automatically actuates Engineered Safety Feature (ESF) systems and their support systems when any of the monitored process variables reach a predetermined setpoint.

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The trip setpoints of the RPS are selected to ensure that Design Basis Events (DBEs) which are expected to occur once or more during the life of the nuclear generating station do not cause the violation of SAFDLs. The reactor trips also help the ESF systems in mitigating the consequences of DBEs which are expected

to occur once during the life of several plants as well as arbitrary combinations of unplanned events and degraded systems that are never expected to occur, to within acceptable limits. Reactor trip is accomplished by de-energizing the Control Element Drive Mechanism (CEDM) coils through the interruption of the CEDM power supply either automatically or manually. The CEDM power supply is a pair of full capacity motor-generator sets. The path to the CEDMs is interrupted by opening the Reactor Trip Switchgear. With the CEDM coils de-energized, the CEAs are released to drop into the core by gravity, rapidly inserting negative reactivity to shut the reactor down. The CEDMs are described in Section 4.2, the specific reactor trips used are described in Section 7.2.

The ESF systems are actuated to minimize the effects of incidents which could occur. Controls are provided for manual actuation of the ESF system. The process variables which automatically actuate the ESF system and the circuitry arrangements for the ESFAS are discussed in Section 7.3. The ESF systems are discussed in Chapter 6. | D

The SAFDL on linear heat rate and DNBR are intended to enforce the principal thermal hydraulic design basis given in Section 4.4.1 (i.e., the avoidance of thermally induced fuel damage during normal steady state operation and during Anticipated Operational Occurrences). | D

3.1.17   CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

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

RESPONSE:

The PPS is designed to provide high functional reliability and in-service testability. The protection system is designed to comply with the requirements of IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," and | D

IEEE 603-1980, "Criteria for Safety Systems for Nuclear Power Generating Stations," and other standards as noted in Section 7.1.2. No credible single failure will result in loss of the protection function. The protection channels are independent with respect to wire routing, sensor mounting, and supply of power.

D

Each channel of the protection system, including the sensors, up to the Reactor Trip Switchgear System (RTSS) and ESFAS actuation devices, is capable of being checked during reactor operation. Process sensors of each channel in the protection systems are checked by comparison of the redundant process sensor values using the discrete indications and alarms on the control room panels as described in Section 7.7.1.3.1. Discrepancies among redundant channel sensors beyond specified limits are alarmed as described in Section 7.7.1.4.3 and Chapter 18.

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The RTSS and ESFAS are described in Chapter 7.

To minimize inadvertent actuation of an ESF system or an inadvertent reactor trip, the protection systems utilize a coincidence of two logics to operate. In addition, the channel being tested is bypassed so that the protection system converts to a two-out-of-three logic while maintaining the coincidence of two. This allows periodic testing and operation of the various protective functions without reducing the availability of the protection systems.

3.1.18 CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

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

RESPONSE:

The protection systems conform to the independence requirements of IEEE 279-1971. Four independent measurement channels, complete with sensors, sensor power supplies, signal conditioning units, and bistable trip functions are provided for each protective parameter monitored by the protection systems except for the CEA position sensors which are two-fold redundant. The measurement channels are provided with a high degree of

independence by separate connection of the channel sensors to the process systems. Refer to Chapter 7 for a more detailed discussion of the protection systems.

Power to the protection system channels is provided by independent vital power supply buses. The power supply systems are discussed in Chapter 8. | D

Functional diversity is incorporated into the system design, to the extent practical, to prevent loss of the protective function. Whenever an RPS trip function is required it is frequently backed up by other trip functions. The ESFAS actuation signals are used to actuate two independent ESF trains. Where it is practical, an ESFAS can be generated by more than one parameter.

The Alternate Protection System augments reactor trip and emergency feedwater actuation by using separate and diverse non-1E trip logic from that used by the Plant Protection System. | D

The design goals are accomplished without excessive complexity by using only four channels for each parameter. This allows for testing and maintenance of a channel without reducing the system to a single channel for trip, which would make the system susceptible to spurious trip or actuation signals.

The protection systems are each functionally tested to ensure satisfactory operation prior to installation in the plant. Environmental and seismic qualifications are also performed utilizing type tests, specific equipment tests, appropriate analyses, or prior operating experience. For further information, refer to Sections 3.10 and 3.11.

3.1.19 CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

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

RESPONSE:

The Plant Protection System trip channels are designed to fail into a safe state or into a state established as acceptable in the event of loss of power supply. A failure is assumed to occur in only one channel (i.e., a single failure). This channel can be placed into bypass which places the RPS/ESFAS local coincidence logic into a two-out-of-three configuration which | D

retains the coincidence of two for trip initiation. Refer to Sections 7.2 and 7.3 for Failure Modes and Effects Analysis information.

A loss of power to CEDM coils will cause the CEAs to insert into the core. Redundancy, channel independence and separation are incorporated into the protection system design to minimize the possibility of the loss of a protective function. The loss of offsite power will cause the standby diesel electric generators to start and energize the ESF trains which have actuation signals present.

3.1.20 CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

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

RESPONSE:

Protection system components and control system components are electrically and functionally isolated from each other. See Sections 7.2, 7.3 and 7.7.1.1.13 for details.

The protection systems are designed so that they can sustain one channel in a tripped condition and one channel bypassed indefinitely and still provide their safety function.

Where control and protection systems have identical sensor input requirements, redundant Class 1E sensors that are used independently by each channel of the protection system may also be used by the control system. For each sensed parameter, the control system monitors all four redundant instrument channels, which are interfaced to the control system via fiber-optic interfaces to ensure electrical independence. Within the control system, signal validation logic is used to detect bypassed or failed sensors, thereby ensuring that they cause no erroneous control system actions. The control system signal validation logic is described in Section 7.7.1.1.13. The design ensures that with a sensor or channel in bypass, another sensor can fail with no resulting control system action. Therefore, with one channel in bypass, the protection system remains in an effective two-out-of-three configuration, meeting the required single failure criteria.

**3.1.21   CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR
REACTIVITY CONTROL MALFUNCTIONS**

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

RESPONSE:

Shutdown of the reactor is accomplished by the opening of the RTSS circuit breakers which interrupts power to the CEDM coils. Actuation of the circuit breakers is independent of any existing control signals.

The protection systems are designed such that SAFDLs are not exceeded for any single malfunction of the reactivity control systems, including the withdrawal of a single full- or part-strength CEA. Analyses of possible reactivity control system malfunctions are discussed in Chapter 15. The various CEA related DBEs for which the protection systems are designed are discussed in Section 7.2. | D

**3.1.22   CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY
AND CAPABILITY**

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

RESPONSE:

Two independent reactivity control systems of different design principles are provided. The first system, using Control Element Assemblies (CEAs), includes a positive means (gravity) for inserting CEAs and is capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including Anticipated Operational Occurrences, SAFDLs are not exceeded. The CEAs can be mechanically driven into the core.

The appropriate margin for stuck rods is provided by assuming in the analyses of anticipated operational occurrences that the highest worth CEA does not fall into the core.

The second system, using neutron absorbing soluble boron, is capable of reliably compensating for the rate of reactivity changes resulting from planned normal power changes (including Xenon burnup) such that SAFDLs are not exceeded. This system is capable of holding the reactor subcritical under cold conditions.

Either system is capable of making the core subcritical from a hot operating condition and holding it subcritical in the hot standby condition.

Either system is able to insert negative reactivity at a rate sufficient to prevent exceeding SAFDLs as the result of a power change (i.e., the positive reactivity added by Xenon burnup).

3.1.23   CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

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

RESPONSE:

Dissolved boron addition capability provided by the Safety Injection System (Chapter 6) in concert with the control rod (CEA) system will be such that under postulated accident conditions (Chapter 15), even with the CEA of highest worth stuck out of the core, adequate reactivity control is available to maintain short- and long-term capability to cool the core.

3.1.24   CRITERION 28 - REACTIVITY LIMITS

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor 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.

RESPONSE:

The bases for Control Element Assembly (CEA) design include ensuring that the reactivity worth of any one CEA is not greater than a preselected maximum value. The CEAs are divided into two sets, a shutdown set and a regulating set, further subdivided into groups as necessary. Administrative procedures and interlocks assure that only one group is withdrawn at a time, and that the regulating groups are withdrawn only after the shutdown groups are fully withdrawn. The regulating groups are programmed to move in sequence and within limits which prevent the rate of reactivity addition and the worth of individual CEAs from exceeding limiting values.

The maximum rate of reactivity addition which may be produced by the Chemical and Volume Control System is too low to induce any significant pressure forces which might rupture the reactor coolant pressure boundary or disturb the reactor vessel internals.

The reactor coolant pressure boundary (Chapter 5) and the reactor internals (Chapter 4) are designed to appropriate codes (refer for instance, to the response to Criterion 14) and will accommodate the static and dynamic loads associated with an inadvertent, sudden release of energy, such as that resulting from a CEA ejection or steam line break (Chapter 15), without rupture and with limited deformation which will not impair the capability of cooling the core.

**3.1.25   CRITERION 29 - PROTECTION AGAINST ANTICIPATED
                     OPERATIONAL OCCURRENCES**

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

RESPONSE:

Plant events, designated in ANSI/ANS 51.1, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," have been carefully considered in the design of the protection and reactivity control systems. Consideration of redundancy, independence and testability in the design, coupled with careful component selection, overall system testing, and

adherence to detailed quality assurance requirements, assure an extremely high probability that safety functions are accomplished in the event of Design Basis Events (DBEs).

Detailed discussions of the protection systems are provided in Chapter 7. Design quality assurance is discussed in Chapter 17. The analysis of DBEs is contained in Chapter 15.

**3.1.26   CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE
BOUNDARY**

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

RESPONSE:

The reactor coolant pressure boundary components are designed, fabricated, erected and tested in accordance with the ASME Code Section III. All components are classified Safety Class 1 or 2, in accordance with the ANSI/ANS 51.1, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," definitions for safety classes and the reactor coolant pressure boundary. Accordingly, they receive all of the quality measures appropriate to that classification. |D

Means are provided for the identification of the source of reactor coolant leakage. These include the detection of leakage to systems connected to the reactor coolant pressure boundary as well as leakage from the boundary into the containment.

Instrumentation is provided to indicate and record makeup flow rate and integrated makeup flow to the primary water system. This instrumentation permits detection of suddenly occurring leaks and those which are gradually increasing. |D

**3.1.27   CRITERION 31 - FRACTURE PREVENTION OF REACTOR
COOLANT PRESSURE BOUNDARY**

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions: (1) the boundary behaves in a nonbrittle manner; and, (2) The probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the

uncertainties in determining: (1) material properties; (2) the effects of irradiation on material properties; (3) residual, steady state, and transient stresses; and, (4) size of flaws.

RESPONSE:

All the reactor coolant pressure boundary components are designed and constructed in accordance with ASME Section III and comply with the test and inspection requirements of these codes. The test and inspection requirements assure that flaw sizes are limited so that the probability of failure by rapid propagation is extremely remote. Particular emphasis is placed on the quality control applied to the reactor vessel on which tests and inspections exceeding ASME code requirements are performed. The tests and inspections performed on the reactor vessel are summarized in Section 5.2.4.1.

Carbon and low alloy steel materials which form part of the pressure boundary are tested in accordance with the requirements of the fracture toughness requirements for materials, ASME Code Section III. Nonductile failure prevention will be ensured by utilizing the appropriate sections of the ASME Code.

Excessive embrittlement of the reactor vessel material due to neutron radiation is prevented by providing an annulus of coolant water between the reactor core and the vessel. In addition, to minimize the effects of irradiation on material toughness properties of core beltline materials, restrictions on upper limits for residual elements that directly influence the RT_{NDT} shift are required by the design specification. Specifically, upper limits are placed on copper, nickel, phosphorous, sulfur, and vanadium. Further, the reactor vessel is forged such that no welds occur in the active core region. | D

The maximum integrated fast neutron flux exposure of the reactor vessel wall opposite the midplane of the core is less than 6.0×10^{19} nvt. This value assumes a sixty-year vessel design life and an eighty percent plant capacity factor. The maximum expected increase in transition temperature is about 140°F. The actual change in material toughness properties due to irradiation will be verified periodically during plant lifetime by a material surveillance program. Based on an initial RT_{NDT} of -20°F, operating restrictions will be applied as necessary to limit vessel stresses. | D
| I

The thermal stresses induced by the injection of cold water into the vessel, following a LOCA, have been examined. Analyses have shown that there is no gross yielding across the vessel wall when using the minimum specified yield strength in the ASME Boiler and Pressure Vessel Code, Section III. | D

**3.1.28   CRITERION 32 - INSPECTION OF REACTOR COOLANT
PRESSURE BOUNDARY**

Components which are part of the reactor coolant pressure boundary shall be designed to permit:

- A. Periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity; and
- B. An appropriate material surveillance program for the reactor pressure vessel.

RESPONSE:

Provisions have been made in the design for inspection, testing, and surveillance of the Reactor Coolant System boundary as required by ASME Boiler and Pressure Vessel Code Section XI. C-E recommends a reactor vessel surveillance program to the owner. The reactor vessel surveillance program capability provided to the site operator conforms with ASTM-E-185-73, "Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Reactor Vessels," as revised in 1982. Sample pieces taken from the same material used in fabrication of the reactor vessel are installed between the core and the vessel inside wall. These samples will be removed and tested by the site operator at intervals during vessel life to provide an indication of the extent of the neutron embrittlement of the vessel wall. Charpy tests will be performed on the samples to develop a Charpy transition curve. By comparison of this curve with the Charpy curve and drop weight tests for specimens taken at the beginning of the vessel life, the change of RT_{NDT} will be determined and operating procedures adjusted as required. See Chapter 5 for further details.

The surveillance program capability provided to the site operator has provisions which comply with the NRC regulation, "Reactor Vessel Material Surveillance Program Requirements," 10 CFR 50, Appendix H, published in the Federal Register in July 1983. The only exception between the recommended surveillance program and the requirements presented in Appendix H is the following:

A. Appendix H, Section II.C.2 - Attachments to the reactor vessel.

In adhering to the requirement of placing the surveillance specimens as close as possible to the reactor vessel wall, the capsule holders are attached to the cladding of the reactor vessel and are not major load-bearing components. By such placement, temperature, flux spectra, and fluence differences between the surveillance specimens and the reactor vessel are minimized, thereby permitting more accurate assessment of the changes in the reactor vessel properties.

3.1.29 CRITERION 33 - REACTOR COOLANT MAKEUP

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

RESPONSE:

Reactor coolant makeup during normal operation is provided by the Chemical and Volume Control System (CVCS). The design incorporates a high degree of functional reliability by provision of redundant components and an alternate path for charging. The charging pumps can be powered from either onsite or offsite power sources, including the emergency diesel generators. The system is described in Section 9.3.4. The CVCS has the capability of replacing the flow loss to the containment due to leaks in small reactor coolant lines such as instrument and sample lines. These lines have 7/32 inch diameter by 1 inch long flow restricting devices. | D

The CVCS is not required to perform any safety related function, such as accident mitigation, or be required to perform a safe shutdown. This does not, however, compromise the "defense in depth" provided by the system as the normal means of maintaining RCS inventory and primary water chemistry. In designing the CVCS as non-safety grade, the following safety functions are performed by dedicated safety systems. Boration and makeup for design | D

basis events will be provided by the Safety Injection System. Pressure control will be provided by the Safety Depressurization System. The Safety Injection System and the Safety Depressurization System are described in further detail in Sections 6.3 and 6.7, respectively. All portions of the CVCS outside of containment have been designed as non-nuclear safety. However, portions of the CVCS which are inside the containment will retain their safety class designation to ensure the integrity of the reactor coolant pressure boundary.

3.1.1.30 CRITERION 34 - RESIDUAL HEAT REMOVAL

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

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

RESPONSE:

Residual heat removal capability is provided by the Shutdown Cooling System for reactor coolant temperatures less than 350°F. For temperatures greater than 350°F, this function is provided by the steam generators. The Emergency Feedwater (EFW) System provides a dedicated, independent, safety-related means of supplying secondary side, quality feedwater to the steam generator(s) for removal of heat and prevention of reactor core uncover. The design incorporates sufficient redundancy, interconnections, leak detection, and isolation capability to ensure that the residual heat removal function can be accomplished, assuming a single active failure. Within appropriate design limits, either system will remove fission product decay heat at a rate such that SAFDLs and the design conditions of the reactor coolant pressure boundary will not be exceeded.

The Shutdown Cooling System and the steam generator auxiliaries are designed to operate either from offsite power or from onsite power sources.

Further discussion is included in Section 5.4.7 for the Shutdown Cooling System and in Chapter 10 for the Steam and Power Conversion System.

3.1.1.31 CRITERION 35 - EMERGENCY CORE COOLING

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

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

RESPONSE:

Emergency core cooling is provided by the Safety Injection System (SIS) (described in Section 6.3). The system is designed to provide cooling water to remove heat at a rate sufficient to maintain the fuel in a coolable geometry and to assure that zirconium-water reaction is limited to a negligible amount (less than one percent). Detailed analysis has been performed, utilizing models complying with 10 CFR 50, Appendix K, "ECCS Evaluation Models," to verify that the system performance is adequate to meet the intent of the "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" of 10 CFR 50.46(b).

The system design includes provisions to assure that the required safety functions are accomplished with either onsite or offsite electrical power, assuming a single failure of any component (qualified as described below). The single failure may be an active failure* during the short-term cooling phase of safety

* An active failure is a malfunction, excluding passive failure, of a component which relies on mechanical movement to complete its intended function upon demand. Check valves which receive regular exercise to ensure operability are treated as passive components. Examples of active failures include the failure of a valve to move to its correct position, or the failure of a pump, fan, or diesel generator to start. Spurious action of a powered component originating within the actuation system or its supporting systems shall be regarded as an active failure, unless specific design features or operating restrictions preclude such spurious action.

injection or an active or limited leakage passive failure* during the long-term cooling phase of safety injection.

Though the SIS is designed to accommodate a limited leakage passive failure during the long-term cooling phase, it does not accommodate arbitrary large leakage passive failures, such as the complete double-ended severance of piping, which are extremely low probability events. The site-specific layout and arrangement will be such that the limited leakage passive failure does not preclude minimum acceptable long-term cooling capability. Where building design is not relied upon to mitigate and contain leakage from the SIS passive failure, suitable automatic isolation and auxiliary equipment must be provided by the site operator, as necessary.

3.1.32   CRITERION 36 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM

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

RESPONSE:

The SIS is designed to facilitate access to all critical components. All pumps, heat exchangers, valves and piping external to the containment structure are readily accessible for periodic inspection to ensure system leak-tight integrity. Valves, piping and tanks inside the containment may be inspected for leak-tightness during plant shutdowns for refueling and maintenance.

Reactor vessel internal structures, reactor coolant piping and water injection nozzles are designed to permit visual inspection for wear due to erosion, corrosion or vibration, and nondestructive inspection techniques where these are applicable and desirable.

Details of the inspection program are described in Chapters 5, 6, and 16.

* A passive failure is defined as the blockage of a process flow path or a breach in the integrity of a component or piping (e.g., a piping failure).

**3.1.33   CRITERION 37 - TESTING OF EMERGENCY CORE COOLING
SYSTEM**

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

RESPONSE:

The SIS is provided with testing capability to demonstrate system and component operability. Testing can be conducted during normal plant operation with the test facilities arranged not to interfere with the performance of the systems or with the initiation of control circuits, as described in Section 6.3 and Chapter 14. | D

3.1.34   CRITERION 38 - CONTAINMENT HEAT REMOVAL

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

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

RESPONSE:

The Containment Spray System consists of two completely independent subsystems. The heat removal capacity of the flow from either containment spray subsystem is adequate to keep the containment pressure and temperature below design conditions for any size break in the RCS piping up to and including a double-ended break of the largest reactor coolant pipe, with an unobstructed discharge from both ends. | D

Borated water is sprayed downward by the system from the upper regions of the containment to cool the atmosphere. Cooling reduces the containment pressure and temperature following a major loss-of-coolant-accident.

Suitable redundancy in components and features is designed into the Containment Spray System to maintain the pressure and temperature conditions below containment design even in the event of a single failure, including the loss of onsite or offsite electrical power.

**3.1.35   CRITERION 39 - INSPECTION OF CONTAINMENT HEAT
REMOVAL SYSTEM**

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

RESPONSE:

All essential equipment of the Containment Spray System is located outside the containment, except for spray headers, nozzles, containment sump, In-containment Refueling Water Storage Tank and associated piping. These components include two containment spray pumps, two containment spray heat exchangers and independent containment spray headers.

The detailed arrangement and layout of system piping, pumps, heat exchangers, and valves will provide the separation, availability, and accessibility required for periodic inspection. Nozzle inspection capability will be provided as well.

**3.1.36   CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL
SYSTEM**

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

RESPONSE:

System piping, valves, pumps, heat exchangers, and other components of the containment heat removal system are arranged so that each component can be tested periodically for operability. Testing can be conducted during normal plant operation with the test facilities arranged not to interfere with the performance of the system or with the initiation of control circuits, as described in Section 6.2.

The performance testing of containment spray pumps is conducted at some time other than refueling. The pumps are aligned to take suction from and return flow to the In-containment Refueling Water Storage Tank (IRWST). Flow and head are recorded by the installed instrumentation.

Heat exchanger operation may be verified during any operating mode by circulating water through the containment spray heat exchanger and back to the IRWST.

Actuator-operated valves can be cycled from the control room, and operation verified by observing control room indication.

Check valves will be tested to ensure that the valves operate properly. These valves include the IRWST check valves and the valves on the inlets and outlets of the containment spray pumps.

3.1.37   CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP

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

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

RESPONSE:

Two systems, namely the Containment Spray and Containment Hydrogen Recombiner Systems, are provided to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment. The Containment Spray and Containment Hydrogen Recombiner Systems are designed with redundancy of vital components so that a single failure does not prevent performance of the safety function coincident with a loss of offsite power. I

The systems are described in detail in Sections 6.2 and 6.5.

**3.1.38   CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE
CLEANUP SYSTEMS**

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

RESPONSE:

The containment atmosphere cleanup systems are designed and located so that they can be inspected periodically as required. Inspection of the Containment Spray System function relative to iodine removal is treated in the response to Criterion 39. I

All major components of the Containment Hydrogen Recombiner System are located outside containment and are readily accessible for periodic inspection. Purge piping and valves located inside containment may be inspected during plant shutdown.

**3.1.39   CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE
CLEANUP**

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

RESPONSE:

Testing of the Containment Spray System shall be conducted to assure structural and leaktight integrity, and operability and performance in accordance with Criterion 40. In addition, performance testing will be conducted on all components of the Containment Spray System. These tests are normally conducted while the plant is operating. System design includes provisions which allow component testing with sufficient safeguards to prevent accidental containment spray. See Sections 6.2.2 and 6.5 for details. I

The Containment Hydrogen Recombiner System is designed to permit periodic testing for structural and leaktight integrity of components and for operability of the system and individual components. Testing may be conducted during normal plant operation or shutdown. See Section 6.2.5 for details.

3.1.40 CRITERION 44 - COOLING WATER

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

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

RESPONSE:

The cooling water systems which function to remove the combined heat load from structures, systems, and components important to safety under normal operating and accident conditions are the Component Cooling Water System and the Station Service Water System. The Component Cooling Water System is a closed loop system which removes heat from nuclear safety related and potentially radioactive systems. The Station Service Water System removes heat from the Component Cooling Water System and transfers it to the atmosphere through cooling ponds. The Station Service Water System is described in Section 9.2.1 and the Component Cooling Water System is described in Section 9.2.2. D

3.1.41 CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM

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

RESPONSE:

The important components are located in accessible areas. These components have suitable manholes, handholes, inspection ports, or other appropriate design and layout features to allow periodic inspection. See Sections 9.2.1 and 9.2.2 for details. | D

3.1.42 CRITERION 46 - TESTING OF COOLING WATER SYSTEM

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

RESPONSE:

The design provides for periodic testing of active components of the cooling water systems for operability and functional performance. | D

Preoperational performance tests of the components are required to be made by the manufacturer. An initial system flow test demonstrates proper functioning of the system. Thereafter, periodic tests ensure that components are functioning properly. | I

Cooling water system valves may be connected to the preferred power source at any time during reactor operation to demonstrate operability. Many active components are operated normally, thereby demonstrating operability. Remotely operated valves are exercised and actuation circuits tested. The automatic actuation circuitry, valves, and pump breakers also may be checked when integrated system tests are performed during a planned cooldown of the Reactor Coolant System. Provisions have been made to permit periodic leakage tests to verify the continued leak-tight integrity of the systems. Refer to Sections 9.2.1 and 9.2.2 for details. | D

3.1.43 CRITERION 50 - CONTAINMENT DESIGN BASIS

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

RESPONSE:

The 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 LOCA up to and including a double-ended rupture of the largest reactor coolant pipe.

The containment structure and Engineered Safety Feature systems have been evaluated for various combinations of energy release. The analysis accounts for system thermal and chemical energy, and for nuclear decay heat. The Safety Injection System is designed such that no single active failure could result in significant metal-water reaction (see Section 6.2.1).

3.1.44 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.

RESPONSE:

The material selected for the containment vessel is carbon steel normalized to refine the grain which results in improved ductility. In addition, the actual mechanical and chemical properties of the material are within the limits of minimum ductility defined in the 1989 ASME Code Material Specifications Part A-SA-537/SA-537M.

The containment vessel is built to Subsection NE of Section III of the ASME Boiler and Pressure Vessel Code. I

The design of the vessel reflects consideration of all ranges of temperature and loading conditions which apply to the vessel during operation, maintenance, testing and postulated accident conditions.

All seam welds in the vessel are 100 percent radiographed, and the acceptance standards of the radiographs ensure that flaws in welds do not exceed the maximum allowed by the ASME Code.

Steady state and transient stresses are calculated in accordance with accepted methods (see Section 3.8). D

**3.1.45   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. D

RESPONSE:

The containment vessel is designed so that integrated leak rate testing can be performed at design pressure after completion and installation of penetrations and equipment in accordance with the requirement of Appendix J of 10 CFR 50 (see Section 6.2.6).

**3.1.46   CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING
AND INSPECTION**

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

RESPONSE:

The absence of insulation on the containment vessel permits periodic inspection of the exposed surfaces of the vessel. The lower portions of the containment vessel are totally encased in concrete and will not be accessible for inspection. It is contemplated that there will be no need for any special in-service surveillance program due to the rigorous design, fabrication, inspection and pressure testing the containment vessel receives prior to operation.

Provisions are made to permit periodic testing at containment design pressure of penetrations which have resilient seals or expansion bellows to allow leak-tightness to be demonstrated (refer to Section 6.2.6).

**3.1.47   CRITERION 54 - PIPING SYSTEMS PENETRATING
CONTAINMENT**

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

RESPONSE:

Piping systems described in CESSAR which penetrate containment are designed to provide the required isolation and testing capabilities. These piping systems are provided with test connections to allow periodic leak detection tests to be performed, in accordance with 10 CFR 50, Appendix J.

The Engineered Safety Features Actuation System circuitry provides the means for testing isolation valve operability.

For a discussion of penetration design, refer to Section 6.2.4, Containment Isolation System.

For additional related discussion, see the responses to General Design Criteria 55, 56, and 57 (Sections 3.1.48 through 3.1.50).

**3.1.48   CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY
PENETRATING CONTAINMENT**

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

- A. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
- B. One automatic isolation valve inside and one locked closed isolation valve outside containment; or
- C. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
- D. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

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

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

RESPONSE:

The reactor coolant system pressure boundary for CESSAR is defined in accordance with ANSI/ANS 51.1 and 10 CFR 50, Section 50.2(v). All reactor coolant pressure boundary lines penetrating containment meet the isolation criteria of GDC 55 using the following basis for specific lines in addition to those noted above.

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- A. Safety injection lines, as shown on Figure 6.3.2-1A, are used to mitigate the consequences of accidents and therefore do not receive an automatic closure signal and are not locked closed. | D
- B. When in the shutdown cooling mode of operation the Shutdown Cooling System is an extension of the reactor coolant pressure boundary. In this mode the system is isolated from the environment by two isolation valves in series.
- C. The charging and seal injection lines shown on Figure 9.3-4 have automatic valves outside the containment which do not receive a closure signal (CIAS). This is because it is desirable to maintain charging and seal injection flow as long as the charging pumps are in operation. | D

3.1.49 CRITERION 56 - PRIMARY CONTAINMENT ISOLATION

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

- A. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
- B. One automatic isolation valve inside and one locked closed isolation valve outside containment; or
- C. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment, or
- D. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

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

RESPONSE:

CESSAR fluid systems comply with the requirements of GDC 56 with the following exceptions:

Lines which connect directly to the containment atmosphere and are used for mitigating the effects of accidents are connected to a closed piping system outside containment, which is isolated from the environment in accordance with the requirements of GDC 55. In addition, the capability for remote double isolation at the containment boundary is provided in accordance with GDC 56.

3.1.50   CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES

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

RESPONSE:

The systems that fall into the category described in GDC 57 comply with containment isolation requirements as specified in the containment isolation system sections of CESSAR.

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3.1.51   CRITERION 60 - CONTROL OF RELEASES OF RADIOACTIVE MATERIAL 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.

RESPONSE:

The sources and expected quantities of radioactive materials produced during normal reactor operation, including anticipated operational occurrences, is presented in Chapter 11. The radioactive waste systems to suitably control the release of these materials in gaseous and liquid effluents and to handle radioactive solid wastes are described in Sections 11.2 through 11.4.

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**3.1.52 CRITERION 61 - FUEL STORAGE AND HANDLING AND
RADIOACTIVITY CONTROL**

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

RESPONSE:

Fuel storage and handling and fuel pool cooling are discussed in Section 9.1. Most of the components and systems in this category are in frequent use and no special testing is required. Those systems and components important to safety that are not normally operating are tested periodically; e.g., the fuel handling equipment (prior to each refueling).

The spent fuel storage racks are located to provide sufficient shielding wear over stored fuel assemblies to limit radiation at the surface of the water to no more than 2.5 mrem/hr during the storage period. The exposure time during refueling is limited so that the integrated dose to operating personnel does not exceed the limits of 10 CFR 20. The accidental release of the maximum activity content of a gas decay tank will not result in doses in excess of 500 mrem whole body. See Chapter 11 for details.

Cooling for the spent fuel pools is designed to prevent damage to fuel in the storage facilities that could result in radioactivity release to the plant operating areas or the plant environs.

**3.1.53 CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL
STORAGE AND HANDLING**

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

RESPONSE:

Both new and spent fuel assemblies are stored in parallel rows designed in accordance with ANSI/ANS 51.1. Normal procedures

require that new fuel be stored in dry air or fully submerged in borated water, and that spent fuel be stored fully submerged in borated water.

The fuel storage and handling system is described in Section 9.1.

Design of the new and spent fuel racks assures a k_{eff} of less than 0.98 for the new and 0.95 for the spent fuel assemblies.

3.1.54   CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

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

RESPONSE:

Instrumentation is provided in the Pool Cooling and Purification System which will detect a loss of residual heat removal capability. Appropriate safety actions are initiated by operator responses. The instrumentation and system relationships are discussed in Section 9.1.

Refer to Section 9.1 for a discussion of Fuel Storage and Handling and to Chapter 11 for a discussion of the area and ventilation system radiation monitoring.

3.1.55   CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

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

RESPONSE:

Provisions are made for monitoring the containment atmosphere, the facility effluent discharge paths, the operating areas within the plant and the facility environs for radioactivity that could be released from normal operation, from anticipated transients, and from an accident.

Some liquid and gaseous effluent will contain radioactive matter. The Waste Management System functions to remove radioactive material from these wastes by filtration and ion exchange prior to discharge.

Liquid wastes are sampled, and if the contained activity meets applicable limits, they may be released with continuous radiation monitoring to the plant dilution flow canal.

Gaseous waste are processed through carbon absorbers to retain and delay radioactive fission gases prior to release. The gas is then monitored during release through the plant vent.

The Condenser Air Removal System discharge is monitored for gaseous activity. The ventilation system discharges are monitored for gaseous activity. Radioactive waste management and monitoring as well as area monitoring are discussed in Chapter 11.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 SEISMIC CLASSIFICATION

Structures, systems, electrical equipment and components which are important to safety and designed to remain functional in the event of a Safe Shutdown Earthquake (SSE) are designated as Seismic Category I.

Seismic Category I structures, systems, and components are those necessary to ensure:

- A. The integrity of the reactor coolant pressure boundary.
- B. The capability to achieve safe shutdown of the reactor and keep it in a safe shutdown condition.
- C. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures in excess of 10 CFR 100 guidelines.

The selection of Category I structures, systems, and components is in accordance with the definition above and the guidance provided by Regulatory Guide 1.29. Individual components in Category I systems are classified by reference to the safety classes assigned in accordance with ANSI/ANS 51.1 (see Section 3.2.2). All components in Safety Classes 1, 2, and 3 are Seismic Category I. |D

Structures, systems and components which do not perform a nuclear safety related function and whose continued function is not required are classified Non-Nuclear Safety (NNS) (see Section 3.2.2). NNS structures, systems and components whose structural failure or interaction could degrade the functioning of a Seismic Category I structure, system, or component to an unacceptable safety level or could result in an incapacitating injury to an occupant of the control room are designated as Seismic Category II and are designed and constructed so that the SSE will not cause such failure in a manner that would adversely affect a safety system. |D

Structures, systems, and equipment which have no enhanced seismic design requirements in addition to those imposed by building codes are designated Non-Seismic (NS).

The seismic category and safety and quality classification of structures, systems, and components within the System 80+ Standard Design are listed in Table 3.2-1 and on the P&IDs (Chapters 5, 6, and 9). Seismic Category I includes all mechanical components within the safety class boundaries and extends to the first seismic restraint beyond the boundary. All fuel racks are also designated as Seismic Category I. Structures, systems, or components whose failure could reduce the performance of a safety function by a Seismic Category I component to an unacceptable safety level are designed to Seismic Category II requirements for structural integrity only or are separated to the extent required to eliminate that possibility. This ensures that any structures, systems, or components that could potentially have a disabling interaction with Seismic Category I structures, systems, or components are either prevented from doing so or are designed to meet Seismic Category I or II structural integrity requirements, depending on the function of the component.

The listing of major electrical components is found in Section 3.11, which also includes safety and quality classifications. Electrical structures, systems, and components not classified as Seismic Category I but whose failure could represent a hazard to the operator or could interfere with the performance of required safety functions of electrical structures, systems and components, are classified as Seismic Category II (Reference 1). Any electrical system or structure or component not in Seismic Category I or II is considered Non-Seismic (see Section 3.10). The use of the Seismic Category II designation for electrical components is limited to non-safety control system components which are designed and documented to maintain structural integrity during an SSE.

For purposes of this discussion, the motors and solenoids used to provide motive power to mechanical components are treated as part of the mechanical component.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATIONS (SAFETY CLASS)

In general, fluid system components important to safety are classified in accordance with ANSI/ANS 51.1 (Reference 2). For purposes of CESSAR, Safety Class 1, 2, 3 and NNS of ANSI/ANS 51.1 are equivalent to Quality Groups A, B, C and D of Regulatory Guide 1.26. The criteria establish safety classes which are used as a guide to the selection of codes, standards, and quality assurance provisions for the design and construction of the components. The safety class designations are also used as a guide to those fluid system components to be classified as Seismic Category I and II (see Section 3.2.1). The Safety Class definitions in ANSI/ANS 51.1 are summarized as follows:

- A. Safety Class 1 (SC-1) applies to pressure-retaining portions and supports of mechanical equipment that form part of the RCPB whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability and whose requirements are within the scope of the ASME Boiler and Pressure Vessel Code, Section III.
- B. Safety Class 2 (SC-2) applies to pressure-retaining portions and supports of primary containment and other mechanical equipment, requirements for which are within the scope of the ASME Boiler and Pressure Vessel Code, Section III, that are not included in SC-1 and are designed and relied upon to accomplish the nuclear safety functions defined in ANSI/ANS 51.1, Section 3.3.1.2.
- C. Safety Class 3 (SC-3) applies to equipment, not included in SC-1 or -2, that is designed and relied upon to accomplish the nuclear safety functions defined in ANSI/ANS 51.1, Section 3.3.1.3.
- D. Non-Nuclear Safety (NNS) applies to equipment that is not in Safety Class 1, 2, or 3. This equipment is not relied upon to perform a nuclear safety function.

The safety classifications of major components which are in the System 80+ design scope are listed in Table 3.2-1 and Section 3.11. Seismic category designations and quality assurance requirements are also included. Small components, such as piping, valves and strainers, are not listed; they may be found by reference to the P&IDs (Chapters 5, 6, and 9) where the exact boundaries are indicated. Valves are listed in Tables 3.2-2.

All pressure containing components in Safety Classes 1, 2, and 3 are designed, manufactured, and tested in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III. Components designated NNS are designed and constructed with appropriate consideration of the intended service using applicable industry codes and standards. The relationship between safety class and code class is shown in Table 3.2-2. A higher code class may be used for a component without changing the safety class or affecting the balance of the system in which it is located.

Fracture toughness requirements are imposed on materials for pressure retaining parts of ASME Class 2 and 3 System 80+ Standard Design components. Test methods, acceptance, and exemption criteria are in conformance with the ASME Code, Section III.

The safety classification system is also used to identify those components to which the requirements of 10 CFR 50, Appendix B,

are applicable. Components in Safety Classes 1, 2, and 3* are designed and manufactured under a rigorous quality assurance program reflecting the requirements of Appendix B, and are designated Quality Class 1. The Quality Class 1 quality assurance program is described in Chapter 17. Components which do not serve a safety related function are designated Quality Class 2. Quality Class 2 components will be designed and manufactured or procured in accordance with the pertinent requirements of the Quality Assurance Program as given in Chapter 17.

The use of the above outlined safety and quality classification systems meets the intent of Regulatory Guide 1.26 and the requirements of 10 CFR 50.55a.

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* With the following exception: the CVCS gas stripper is Safety Class 3, Quality Class 2, however, pressure retaining portions meet rules applicable to ASME Code Class 3 components. See Table 3.2-1.

REFERENCES FOR SECTION 3.2

1. "Seismic Qualification of C-E Instrumentation Equipment,"
Combustion Engineering, Inc., CENPD-182, Revision 1, May
1977.
2. "Nuclear Safety Criteria for the Design of Stationary
Pressurized Water Reactor Plants," ANSI/ANS 51.1, 1983.

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TABLE 3.2-1

(Sheet 1 of 17)

**CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS**

<u>Component Identification</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
Reactor Coolant System			
* Reactor Vessel	1	I	1
* Steam Generators (primary/secondary)	1/2 (1)	I	1
* Pressurizer	1	I	1
* Reactor Coolant Pumps (2) (3) (9)	1	I	1
Piping within Reactor Coolant Pressure Boundary (5)	1/2 (4)	I	1
Control Element Drive Mechanisms	(6)	(6)	1
Core Support Structures (7)	3	I	1
Fuel Assemblies (8)	2	I	1
Control Element Assemblies (8)	3	I	1
Closure Head Lift Rig	NNS	II (10)	2
Heated Junction Thermocouple Probe Assembly	1/3 (12)	I	1
HJTC Pressure Housing	1	I	1
ICI Cable Tray Support Frame	3	I	1
ICI Holding Frame	NNS	NS	2
ICI Guide Tubes	1	I	1
ICI Guide Tube Supports	1	I	1
ICI Seal Housing	1	I	1
ICI Seal Table	1	I	1
Safety Injection System			
* Safety Injection Pumps	2	I	1
* Shutdown Cooling Heat Exchangers	2/3 (1)	I	1
* Safety Injection Tanks	2	I	1
* Shutdown Cooling Pumps	2	I	1
* Containment Spray Pumps	2	I	1
* Containment Spray Heat Exchangers	2/3 (1)	I	1
* IRWST	2	I	1
* Shutdown Cooling Mini-flow Heat Exchanger	2/3 (1)	I	1
* Containment Spray Mini-flow Heat Exchanger	2/3 (1)	I	1

Footnotes to this table are given at the end of the table.

* Including component supports down to (but not including) embedments.

TABLE 3.2-1 (Cont'd)

(Sheet 2 of 17)

**CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS**

<u>Component Identification</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
Chemical and Volume Control System			
* Regenerative Heat Exchanger	2	I	1
* Letdown Heat Exchanger	2	I	1
* Seal Injection Heat Exchanger	2	I	1
* Purification Ion Exchangers	2	I	1
* Deborating Ion Exchanger	2	I	1
* Volume Control Tank	2	I	1
* Chemical Addition Package	NNS	NS	2
* Boric Acid Batching Tank	NNS	NS	2
* Charging Pumps	2	I	1
* Boric Acid Makeup Pumps	3	I	1
* Reactor Makeup Water Pumps	NNS	NS	2
* Boric Acid Concentrator	NNS	NS	2
* Pre-holdup Ion Exchanger	3	I	1
* Mini-flow Heat Exchanger	2	I	1
* Boric Acid Condensate Ion Exchanger	NNS	NS	2
* Reactor Drain Pumps	3	I	1
* Holdup Pumps	NNS	NS	2
* Reactor Drain Tank	NNS	NS	2
* Holdup Tank	NNS	NS	2
* Equipment Drain Tank	3	I	1
* Reactor Makeup Water Tank	NNS	NS	2
* Gas Stripper	3	I	1
* Purification Filters	2	I	1
* Reactor Drain Filter	3	I	1
* Seal Injection Filters	2	I	1
* Reactor Makeup Filter	NNS	NS	2
* Boric Acid Filter	3	I	1
* Letdown Strainer	2	I	1
* Pre-holdup Strainer	3	I	1
* Boric Acid Condensate IX Strainer	NNS	NS	2
* Ion Exchanger Drain Header Strainer	NNS	NS	2
* Boric Acid Batching Strainer	NNS	NS	2
* Chemical Addition Strainer	NNS	NS	2
* Boric Acid Storage Tank	2	I	1

TABLE 3.2-1 (Cont'd)

(Sheet 3 of 17)

**CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS**

<u>Component Identification</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>	
Emergency Feedwater System				
* Motor-Driven Emergency Feedwater Pumps	3	I	1	I
* Steam-Driven Emergency Feedwater Pumps	3	I	1	
* Emergency Feedwater Pump Turbines	3	I	1	
* Emergency Feedwater Storage Tanks	3	I	1	
Fuel Handling System				
Refueling Machine	NNS	II	2	D
Fuel Transfer System	NNS	II	2	
1. Transfer Carriage	NNS	II	2	
2. Upending Machine	NNS	II	2	
3. Hydraulic Power Unit	NNS	II	2	
Fuel Transfer Tube, Valve, Stand	NNS	II	2	
CEA Change Platform	NNS	II	2	
Long and Short Fuel Handling Tools	NNS	NS	2	
Upper Guide Structure Lifting Rig	NNS	II (11)	2	
Core Barrel Lifting Rig	NNS	II (11)	2	
Spent Fuel Handling Machine	NNS	II	2	
New Fuel Elevator	NNS	II	2	
Underwater Television	NNS	NS	2	
Refueling Pool Seal	NNS	NS	2	
In-Core Instrumentation and CEA Cutter	NNS	NS	2	
Extension Shaft Uncoupling Tool	NNS	NS	2	
Fuel Transfer Tube Blind Flange	2	I	1	E
CEA Handling Tools	NNS	NS	2	
ICI Insertion and Removal Tools	NNS	NS	2	
Spent Fuel Racks	NNS	I	2	
New Fuel Racks	NNS	I	2	
Condensate and Feedwater System (13)				
Condensate Pumps	NNS	NS	2	I
Feedwater Pumps	NNS	NS	2	
Feedwater Pump Controllers	NNS	NS	2	
Feedwater Booster Pumps	NNS	NS	2	
Feedwater Startup Pumps	NNS	NS	2	
Low Pressure Feedwater Heaters	NNS	NS	2	
High Pressure Feedwater Heaters	NNS	NS	2	
Deaerator	NNS	NS	2	

TABLE 3.2-1 (Cont'd)**(Sheet 4 of 17)****CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS**

<u>Component Identification</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
Main Condenser System			
Main Condenser	NNS	NS	2
Condensate Storage System			
Condensate Storage Tanks	NNS	NS	2
Condensate Storage Tank Recycle Pump	NNS	NS	2
Condensate Drain Tanks	NNS	NS	2
Condensate Drain Tank Transfer Pump	NNS	NS	2
Condensate Storage Feedwater Startup Pump	NNS	NS	2
Condensate Cleanup System			
Polishers/Demineralizers	NNS	NS	2
Resin Traps	NNS	NS	2
Main Vacuum System			
Vacuum Pumps	NNS	NS	2
Steam Jet Air Ejectors	NNS	NS	2
Steam Jet Air Ejector Condenser	NNS	NS	2
Demineralized Water Makeup System			
Demineralizer Makeup Water Pump	NNS	NS	2
Demineralizers	NNS	NS	2
Vacuum Degasifier	NNS	NS	2
Demineralized Water Storage Tank	NNS	NS	2
Vacuum Pumps	NNS	NS	2
Filters	NNS	NS	2
Recirculating Pumps	NNS	NS	2
Extraction Steam System	NNS	NS	2
Heater Vents	NNS	NS	2

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TABLE 3.2-1 (Cont'd)**(Sheet 5 of 17)****CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS**

Component Identification	Safety Class	Seismic Category	Quality Class
Chemical Feed System			
Chemical Addition Tanks	NNS	NS	2
Chemical Addition Pumps	NNS	NS	2
Turbine Generator System			
Turbine Generator	NNS	NS	2
High Pressure Turbine	NNS	NS	2
Low Pressure Turbines	NNS	NS	2
Generator	NNS	NS	2
Moisture Separators	NNS	NS	2
Steam Reheaters	NNS	NS	2
Turbine Bypass System	NNS	NS	2
Turbine Gland Sealing System			
Gland Seal Condenser	NNS	NS	2
Gland Seal Regulator	NNS	NS	2
Turbine Lube Oil System			
Centrifugal Oil Pumps	NNS	NS	2
Booster Pump	NNS	NS	2
Oil Tank	NNS	NS	2
Oil Turbine	NNS	NS	2
Oil Coolers	NNS	NS	2
Turbine Control System			
EHC Pumps	NNS	NS	2
EHC Coolers	NNS	NS	2
EHC Sumps	NNS	NS	2
Turbine Generator Cooling System			
Hydrogen Coolers	NNS	NS	2
Liquid Waste Management System			
Waste Collection Tanks	NNS	NS	2
Waste Sample Tanks	NNS	NS	2
Process Pumps	NNS	NS	2
Process Demineralizers	NNS	NS	2
Process Filters	NNS	NS	2

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TABLE 3.2-1 (Cont'd)**(Sheet 6 of 17)****CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS**

<u>Component Identification</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
Gaseous Waste Management System			
Gas Dryers	NNS	NS	2
Charcoal Beds	NNS	NS	2
Solid Waste Management System			
HIC Fill/Dewatering Head	NNS	NS	2
Slurry Pump	NNS	NS	2
Radwaste Building Crane	NNS	NS	2
Dry Solids Compactor	NNS	NS	2
Heater Drain System			
Reheater Drain Tanks	NNS	NS	2
Moisture Separator Drain Tanks	NNS	NS	2
Heater Drain Tank	NNS	NS	2
Heater Drain Pumps	NNS	NS	2
Process and Effluent Radiation Monitoring System			
Gaseous Process and Effluent Monitors			
Process Vent	NNS	NS	2
Process Gas	NNS	NS	2
Ventilation Vent	NNS	NS	2
Containment Purge Exhaust	NNS	NS	2
Condenser Air Ejector	NNS	NS	2
Reactor Building Annulus	NNS	II	2
Subsphere Ventilation	NNS	NS	2
Liquid Process and Effluent Monitors			
Component Cooling Water	NNS	NS	2
Liquid Waste Discharge	NNS	NS	2
Plant Discharge Line	NNS	NS	2
Station Service Water	NNS	NS	2
Reactor Coolant Gross Activity	NNS	NS	2
Turbine Building Drains	NNS	NS	2
Steam Generator Blowdown	NNS	NS	2

TABLE 3.2-1 (Cont'd)

(Sheet 7 of 17)

**CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS**

Component Identification	Safety Class	Seismic Category	Quality Class
Airborne Radiation Monitors			
Containment Atmosphere	NNS	NS	2
Nuclear Annex	NNS	NS	2
Radwaste Building	NNS	NS	2
Fuel Building	3	I	1
Ventilation Systems Multisampler	NNS	NS	2
Control Room Intake (A&B)	3	I	1
Area Radiation Monitors	NNS	NS	2
Special Purpose Area Monitors			
Main Steam Line	NNS	II	2
Purification Filter	NNS	NS	2
Containment Area High Radiation	3	I	1
Primary Coolant	3	I	1
Containment Isolation System	2	I	1
Component Cooling Water System (14)			
CCWS Heat Exchangers	3	I	1
CCWS Pumps	3	I	1
CCWS Surge Tanks	3	I	1
CCWS Sump Pumps	NNS	NS	2
CCWS Chemical Addition Tank	NNS	NS	2
CCWS Heat Exchanger Building Sump Pumps	NNS	NS	2
Pool Cooling and Purification System			
Spent Fuel Pool Cooling System			
Spent Fuel Pool Cooling Pumps	3	I	1
Spent Fuel Pool Cooling Heat Exchangers	3	I	1
Pool Purification System			
Pool Purification Pumps	NNS	NS	2
Pool Strainers	NNS	NS	2
Pool Demineralizers	NNS	NS	2
Pool Filters	NNS	NS	2
Spent Fuel Pool Skimmer	NNS	NS	2

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TABLE 3.2-1 (Cont'd)**(Sheet 8 of 17)****CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS**

<u>Component Identification</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
Sampling System			
Process Sample System Heat Exchangers	NNS	NS	2
Station Service Water System			
SSWS Pumps	3	I	1
SSWS Strainers	3	I	1
SSWS Sump Pumps	3	I	1
SSWS Traveling Screen	3	I	1
Turbine Building Service Water System			
TBSWS Pumps	NNS	NS	2
Turbine Building Cooling Water System			
TBCWS Heat Exchangers	NNS	NS	2
TBCWS Pumps	NNS	NS	2
TBCWS Surge Tank	NNS	NS	2
TBCWS Chemical Addition Tank	NNS	NS	2
Chilled Water Systems			
Essential Chilled Water System			
ECW Refrigeration Units	3	I	1
ECW Circulation Pumps	3	I	1
ECW Compression Tanks	3	I	1
Normal Chilled Water System (15)			
NCW Refrigeration Units	NNS	I	2
NCW Circulation Pumps	NNS	I	2
NCW Compression Tanks	NNS	I	2
NCW Air Separators	NNS	I	2
NCW Heat Exchangers	3	I	2
Condenser Circulating Water System			
Condenser Circulating Water Pumps	NNS	NS	2
Condenser Circulating Water Cooling Towers	NNS	NS	2

TABLE 3.2-1 (Cont'd)**(Sheet 9 of 17)****CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS**

Component Identification	Safety Class	Seismic Category	Quality Class
Compressed Air Systems			
Instrument Air System			
Air Compressors	NNS	NS	2
Aftercoolers	NNS	NS	2
Moisture Separators	NNS	NS	2
Air Receivers	NNS	NS	2
Desiccant Air Dryers	NNS	NS	2
Air Filters	NNS	NS	2
Station Air System			
Air Compressors	NNS	NS	2
Aftercoolers	NNS	NS	2
Moisture Separators	NNS	NS	2
Air Receivers	NNS	NS	2
Breathing Air System			
Air Compressors	NNS	NS	2
Aftercoolers	NNS	NS	2
Air Filters	NNS	NS	2
Air Receivers	NNS	NS	2
Air Dryer	NNS	NS	2
Compressed Gas Systems			
High Pressure Gas Cylinders	NNS	NS	2
Pressure Regulators	NNS	NS	2
Leak Detection Systems	NNS	NS	2
Liquid Nitrogen Evaporators	NNS	NS	2
Fire Protection System (16)			
Fire Pumps	NNS	I	2
FPS Tank	NNS	I	2
Water Spray Systems (Deluge and Sprinkler)	NNS	II	2
Detection/Alarm System	NNS	NS	2
Hose Systems/Standpipes	NNS	I	2
Air Packs	NNS	NS	2
Portable Fire Extinguishers	NNS	I	2

TABLE 3.2-1 (Cont'd)**(Sheet 10 of 17)****CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS**

<u>Component Identification</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
Diesel Generator Systems			
DG Fuel Oil System (17)			
Fuel Oil Storage Tank	3	I	1
Recirculation Pump	3	I	1
Booster Pump	3	I	1
Fuel Oil Day Tank	3	I	1
Fuel Oil Transfer Pump	3	I	1
DG Cooling Water System			
DGCW Surge Tank	3	I	1
DGCW Circulation Pump	3	I	1
DGCW Keep Warm Pump	3	I	1
3-way Thermostatic Control Valve	3	I	1
DGCW Jacket Water Cooler	3	I	1
DG Starting Air System (18)			
Compressors	3	I	1
Aftercoolers	3	I	1
Moisture Separator	3	I	1
Filter/Dryer Units	3	I	1
Air Receivers	3	I	1
DG Lube Oil System (19)			
Lube Oil Cooler	3	I	1
Oil Transfer Pumps	3	I	1
Prelube Oil Pump	3	I	1
Lube Oil Sump Tank Heaters	3	I	1
Clean and Used Lube Oil Storage Tanks	3	I	1
DG Intake and Exhaust System			
Turbocharger	3	I	1
Aftercooler	3	I	1
Silencers and Air Filters	3	I	1
Equipment and Floor Drainage System			
Reactor Building Subsphere Sump Pumps	3	I	1
DG Building Sump Pumps	3	I	1
Other Floor Drain Sump Pumps	NNS	NS	2

TABLE 3.2-1 (Cont'd)

(Sheet 11 of 17)

**CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS**

<u>Component Identification</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
Control Building Ventilation System			
Main Control Room Air Handling System			
Air Handling Units w/Filters	3	I	1
Fans, Ductwork	3	I	1
Water-cooling Coils	3	I	1
Heating Coils	3	I	1
Technical Support Center Air Handling System			
Air Handling Units	NNS	NS	2
Fans, Ductwork	NNS	NS	2
Filters	NNS	NS	2
Computer Room Air Handling System			
Air Handling Units	NNS	NS	2
Fans, Ductwork	NNS	NS	2
Vital Instrumentation and Equipment Rooms (inc. Battery Rooms)	3	I	1
Balance of Building Air Handling System			
Filters	NNS	NS	2
Water Cooling Coils	NNS	NS	2
Fans, Ductwork	NNS	NS	2
Fuel Building Ventilation System			
Cooling Coil	NNS	NS	2
Heating Coil	NNS	NS	2
Air Handling Unit w/Filter	NNS	NS	2
Ductwork, Supply	NNS	NS	2
Exhaust System Filter Train	3	I	1
Exhaust System Fans	3	I	1
Exhaust System Dampers	3	I	1
Ductwork, Exhaust	3	I	1
Nuclear Annex and Radwaste Building Ventilation System (20)			
Supply Air Handling Units	NNS	NS	2
Ductwork, Supply	NNS	NS	2
Cooling Coils	NNS	NS	2
Particulate Exhaust Filter Units	NNS	NS	2
Fans, Ductwork	NNS	NS	2

TABLE 3.2-1 (Cont'd)

(Sheet 12 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

<u>Component Identification</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
Reactor Building Subsphere Ventilation System			
Exhaust Fans	3	I	1
Cooling Coils and Heating Coils	3	I	1
Exhaust System Filter Train	3	I	1
Ductwork, Exhaust	3	I	1
Supply Fans	NNS	NS	2
Supply Air Handling Units	NNS	NS	2
Ductwork, Supply	NNS	NS	2
Diesel Building Ventilation System			
Supply/Exhaust Fans	3	I	1
Ductwork	3	I	1
Dampers	3	I	1
Annulus Ventilation System			
Filter Trains	3	I	1
Fans, Ductwork	3	I	1
Dampers	3	I	1
Containment Purge Ventilation System			
Water Cooling Coil	NNS	I	2
Heating Coil	NNS	I	2
Supply and Exhaust Fans	NNS	I	2
Dampers	NNS	I	2
Filter Trains	NNS	I	2
Containment Cooling and Ventilation System			
Containment Cooling Subsystem	NNS	I	2
Control Element Drive Mechanism Cooling Subsystem	NNS	I	2
Containment Air Cleanup System	NNS	I	2
Cavity Cooling Subsystem	NNS	I	2
Ductwork	NNS	I	2

TABLE 3.2-1 (Cont'd)**(Sheet 13 of 17)****CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS**

<u>Component Identification</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
Turbine Building Ventilation System			
Ventilating Fans	NNS	NS	2
Intake Dampers	NNS	NS	2
Exhausters	NNS	NS	2
Station Service Water Pump Structure Ventilation System			
Vane-Axial Supply Fans	3	I	1
Dampers/Check Dampers	3	I	1
Ductwork	3	I	1
Main Steam Supply System (21)	2	I	1
Combustible Gas Control System			
Hydrogen Recombiners	2	I	1
Hydrogen Analyzers	2	I	1
Hydrogen Recombiner Control Panel	2	I	1
Steam Generator Blowdown System (22)			
Flash Tank	NNS	NS	2
Blowdown Heat Exchanger	NNS	NS	2
Blowdown Filter	NNS	NS	2
Blowdown Demineralizers	NNS	NS	2
Steam Generator Wet Layup System (22)			
Wet Layup Pumps	NNS	NS	2
Wet Layup Heat Exchangers	NNS	NS	2
Hydrogen Mitigation System			
Hydrogen Igniters	NNS	I	2

TABLE 3.2-1 (Cont'd)

(Sheet 14 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

<u>Component Identification</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
Structures			
Containment Shield Building		I	
Steel Containment		I	
Internal Structure		I	
Nuclear System Annex		I	
Station Service Water System Pump Structure		I	
Turbine Building		NS	
Radwaste Building		NS	
Component Cooling Water Hx Building		I	
Nuclear Service Water Pump Building		I	I
Diesel Generator Building		I	
Main Steam Valve House		I	
Service Building		NS	
Administration Building		NS	
Guard House		NS	
Warehouse		NS	
Fire Pump House		NS	

TABLE 3.2-1 (Cont'd)**(Sheet 15 of 17)****CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS**

- NOTES:**
- (1) Two safety classes are used for heat exchangers to distinguish primary and secondary sides where they are different.
  - (2) Loss of cooling water and/or seal water service to the reactor coolant pumps (RCPs) may require stopping the pumps. However, the continuous operation of the pumps is not required during or following an SSE. The auxiliaries are therefore not necessarily Safety Class 3 or Seismic Category I. Provision for cooling water to the pump bearing oil cooler and pump motor air cooler will not comply with the requirements of Regulatory Guide 1.29 (see Section 5.4.1.3).
  - (3) Only those structural portions of the RCPs which are necessary to assure the integrity of the reactor coolant pressure boundary are Safety Class 1.
  - (4) Safety class of piping within the reactor coolant pressure boundary (as defined in 10 CFR 50) is selected in accordance with the ANSI/ANS 51.1 criteria identified in Section 3.2.2. For purposes of CESSAR, Safety Class 1, 2, 3, and NNS of ANSI/ANS 51.1 are equivalent to Quality Groups A, B, C, and D of Regulatory Guide 1.26. D
  - (5) Flow restricting orifices are provided in the nozzles for RCS sampling lines, pressurizer level and pressure instruments, RCP differential pressure instrument lines, SIS pressure instrument lines, RCP seal pressure instrument lines, the charging line differential pressure instrument line, and the SIS hot leg injection pressure instrument lines, to limit flow in the event of a break downstream of the nozzle. The orifice size, 7/32-inch diameter and 1-inch long, precludes exceeding fuel design limits while utilizing minimum makeup rates. This permits an orderly shutdown in the event of a downstream break in accordance with General Design Criterion 33 (see Section 3.1.29). A reduction may, therefore, be made in the safety classification of lines downstream of the orifice. I
  - (6) The pressure boundary housing for this component is a reactor vessel appurtenance and is Safety Class 1 and Seismic Category I, as described in Section 3.9.4.3.

TABLE 3.2-1 (Cont'd)**(Sheet 16 of 17)****CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS****NOTES:**
(Cont'd)

- (7) Core support structures are designed to the criteria described in Section 3.9.5.4.
- (8) CEA and fuel assemblies are designed to the criteria described in Section 4.2.
- (9) Reactor coolant pump auxiliary components required for lubrication and cooling of pump seals and thrust bearings are Quality Class 2.
- (10) Except Lifting Frame Assembly, which is NS.
- (11) During normal plant operation only.
- (12) Safety Class 1 for pressure boundary; Safety Class 3 for electrical portion of system.
- (13) The piping, valves, and associated supports/restraints of the Main Feedwater System from (and including) the Main Feedwater Isolation Valves to the steam generator feed nozzles are Safety Class 2, Seismic Category I, Quality Class 1; the remainder is ANSI/ASME B31.1.
- (14) Non-safety Cooling Headers are Safety Class NNS, Seismic Category II, and Quality Class 2.
- (15) The Normal Chilled Water System serves no safety function. Portions of the system which are located in non-safety related areas are classed as non-seismic.
- (16) Portions of the Fire Protection System piping and valves which are not in safety-related areas of the plant are designed as non-seismic.
- (17) Fuel Oil Recirculation System and storage tank fill line strainer are classed as non-nuclear safety.
- (18) The Starting Air System is Safety Class NNS from the starting air compressor through the desiccant drying towers, and Safety Class 3 from the starting air receiver tank inlet check valve to the engine connections.

TABLE 3.2-1 (Cont'd)

(Sheet 17 of 17)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

NOTES:
(Cont'd)

- (19) The Clean and Used Oil Transfer System is Safety Class NNS.
- (20) Mechanical Equipment Room cooling components are Safety Class 3, Seismic Category I, and Quality Class 1.
- (21) The piping, valves, and associated supports/restraints of the Main Steam System from each steam generator to (and including) the Main Steam Isolation Valves are Safety Class 2, Seismic Category I, Quality Class 1; the remainder is ANSI/ASME B31.1.
- (22) Piping is Safety Class 2 from the Steam Generators through the Containment Isolation Valves.

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TABLE 3.2-2

(Sheet 1 of 15)

SAFETY CLASS 1, 2 & 3 VALVES

<u>Component Identification</u>	<u>Location/ Description</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
Reactor Coolant System (RCS) (1)				
RC-212	Reactor vessel vent	2	I	1
RC-214	Refueling level indicator (Hot leg)	1	I	1
RC-215, 216, 232, 332, 233, 333, 234, 334, 235, 335, 310, 311, 312, 313, 314, 315, 316, 317	RCS drains	1	I	1
RC-248, 249, 252, 253, 256, 257, 260, 261	Reactor coolant pump (RCP) pressure differential	2	I	1
RC-208, 209, 218, 219, 220	Pressurizer (Pzr) level indicator	2	I	1
RC-204, 205, 206, 207	Pzr pressure indicator	2	I	1
RC-239	Pressurizer refueling	1	I	1
RC-200, 201, 202, 203	Pressurizer safety	1	I	1
RC-240, 241, 442, 443, 236, 237	Pressurizer spray line	1	I	1
RC-100E, 100F	Pzr spray line control	1	I	1
RC-244	Pzr spray line check	1	I	1
RC-210, 213, 238	Sample system	2	I	1
RC-211, 403	Reactor vessel closure head leakoff	2	I	1
RC-217	O-ring leakoff pressure indicator	2	I	1
RC-265, 266	Mid-loop operating connection to IRWST	2	I	1
RC-268, 269	Mid-loop operating connection to pzr	1	I	1
RC-270, 271, 272, 273, 274, 275, 276, 277, 278, 279, 280, 281, 282, 283, 284, 285	Steam generator differential pressure	2	I	1
RC-292, 293, 294, 295, 296, 297, 298, 299	RCS pressure differential	2	I	1
RC-752, 753, 754, 755	RCP seal housing drain	1	I	1
RC-712, 713, 714, 715	RCP vent	2	I	1
RC-446, 447, 448, 449, 450, 451, 452, 453	RCP HP cooler	1	I	1

TABLE 3.2-2 (Cont'd)**(Sheet 2 of 15)****SAFETY CLASS 1, 2 & 3 VALVES**

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
RC-868, 869, 870, 871, 700, 701, 702, 703	RCP filter drain	1	I	1
RC-724, 725, 726, 727, 736, 737, 738, 739	RCP seal cooler pressure	2	I	1
RC-430, 431, 432, 433, 344, 345, 346, 347	RCP controlled bleedoff	2	I	1
RC-380, 381, 382, 383	RCP vapor seal pressure indicator	2	I	1
Main Steam & Feedwater System (MS&FW) (1)				
SG-105, 106, 107, 108	ADV block valve	2	I	1
SG-130, 135, 172, 175	Downcomer isolation	2	I	1
SG-132, 137, 174, 177	Economizer isolation	2	I	1
SG-140, 141, 150, 151	Main steam isolation	2	I	1
SG-169, 183	MSIV bypass valve	2	I	1
SG-178, 179, 184, 185	Atmospheric dump valve	2	I	1
SG-552, 553	SG test connection	2	I	1
SG-554, 555, 556, 557, 558, 559, 560, 561, 572, 573, 574, 575, 576, 577, 578, 579, 691, 692, 694, 695	Main steam safety valve	2	I	1
SG-567, 598, 599, 612, 650, 651	Economizer check valve	2	I	1
SG-586, 587, 605, 609	Downcomer drain valve	2	I	1
SG-603, 611, 661, 665	SG purge connection	2	I	1
SG-608, 644	Economizer drain valve	2	I	1
SG-613, 614, 615, 616, 617, 618, 621, 622, 623, 624, 625, 626, 627, 628, 629, 630, 631, 632, 635, 636, 637, 638, 639, 640	SG level indication	2	I	1
SG-619, 620, 633, 634, 658, 659, 662, 663	Main steam flow indication	2	I	1
SG-642, 643, 652, 653	Downcomer check valve	2	I	1
SG-684, 685, 686, 687	Main steam purge	2	I	1

TABLE 3.2-2 (Cont'd)**(Sheet 3 of 15)****SAFETY CLASS 1, 2 & 3 VALVES**

<u>Component Identification</u>	<u>Location/ Description</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
Chemical and Volume Control System (CVCS) (1)				
CH-101	Letdown check to VCT	2	I	1
CH-103	VCT pressure indicator	2	I	1
CH-104	VCT vent isolation	2	I	1
CH-110P, 110Q	Letdown control valve	2	I	1
CH-112	VCT gas supply check	2	I	1
CH-113, 114	VCT level indicator	2	I	1
CH-115	VCT to EDT relief	2	I	1
CH-116	VCT to RDH isolation	2	I	1
CH-117	VCT local sample	2	I	1
CH-118	VCT discharge check	2	I	1
CH-124	BAST supply isolation	2	I	1
CH-126	BABT to BAST isolation	3	I	1
CH-127	BAC line to BAST check	3	I	1
CH-128, 129	BAST level indicator	2	I	1
CH-130	BAMP recirc isolation	3	I	1
CH-131	Boric acid filter D/P	3	I	1
CH-134	BAMP to DRDH isolation	3	I	1
CH-135	BAST level indicator	2	I	1
CH-139	Gas stripper to VCT	2	I	1
CH-143	BAMP suction isolation	3	I	1
CH-144	BAST to PCPS isolation	3	I	1
CH-145	BAMP suction isolation	3	I	1
CH-146, 147	BAMP discharge pressure	3	I	1
CH-150	VCT level indication	2	I	1
CH-152, 153	BAMP discharge isolation	3	I	1
CH-154, 155	BAMP discharge check	3	I	1
CH-156	BAST level indicator	2	I	1
CH-160	VCT level indication	2	I	1
CH-161	Boric acid filter isolation	3	I	1
CH-164	Boric acid filter bypass	3	I	1
CH-165	Boric acid filter D/P	3	I	1
CH-166	Boric acid filter	3	I	1
CH-172	F-210Y isolation	3	I	1
CH-174	Boric acid makeup cross-connect	3	I	1
CH-176	BAMP local sample	3	I	1
CH-177	Boric acid to charging pump suction check	2	I	1

TABLE 3.2-2 (Cont'd)**(Sheet 4 of 15)****SAFETY CLASS 1, 2 & 3 VALVES**

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
CH-179	RMW line to charging pump suction check	2	I	1
CH-184	RMW line to VCT check	3	I	1
CH-188	RMW to VCT check	2	I	1
CH-190, 191	BAST gravity feed check	2	I	1
CH-192	BAMP to BAST recirc	3	I	1
CH-198	RCP controlled bleedoff	2	I	1
CH-199	RCP controlled bleedoff to RDT relief	2	I	1
CH-205	Auxiliary spray control	1	I	1
CH-208	Charging backpressure	1	I	1
CH-210Y	Boric acid flow control	3	I	1
CH-231	Seal injection isolation	2	I	1
CH-241, 242, 243, 244	Seal injection flow control	2	I	1
CH-255	Seal injection isolation	2	I	1
CH-300	RCP bleedoff pressure	2	I	1
CH-301	Letdown orifice bypass	2	I	1
CH-304	SCS Purification check	2	I	1
CH-307	SCS Purification isolation	2	I	1
CH-308	Letdown heat exchanger vent	3	I	1
CH-330	BAMP line to HT isolation	3	I	1
CH-344	Letdown flow indicator	2	I	1
CH-346	Letdown pressure control isolation	2	I	1
CH-347, 348, 349, 350	Letdown control valve iso.	2	I	1
CH-351	Letdown flow indicator	2	I	1
CH-353	Sampling system isolation	2	I	1
CH-354	Letdown to EDT relief	2	I	1
CH-355	Letdown filter bypass	2	I	1
CH-356, 357	Letdown filter D/P iso.	2	I	1
CH-358	Letdown filter isolation	2	I	1
CH-359	Letdown filter vent	2	I	1
CH-360	Letdown filter isolation	2	I	1
CH-361	Letdown to DRDH isolation	2	I	1
CH-366	Letdown filter vent	2	I	1
CH-369	Ion exchanger isolation	2	I	1
CH-370	IX inlet check	2	I	1
CH-371	IX vent to GWMS	2	I	1
CH-372	IX resin fill isolation	2	I	1
CH-373	Letdown filter isolation	2	I	1

TABLE 3.2-2 (Cont'd)

(Sheet 5 of 15)

SAFETY CLASS 1, 2 & 3 VALVES

<u>Component Identification</u>	<u>Location/ Description</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
CH-374	IX isolation	2	I	1
CH-375	Letdown to DRDH	2	I	1
CH-376	Letdown filter	2	I	1
CH-378	IX isolation	2	I	1
CH-379	RSSH to IX isolation	2	I	1
CH-380	IX to SWMS isolation	2	I	1
CH-381	IX bypass	2	I	1
CH-382, 383	IX isolation	2	I	1
CH-384	IX check	2	I	1
CH-385	IX bypass	2	I	1
CH-386	IX vent to GWMS	2	I	1
CH-387	IX resin fill isolation	2	I	1
CH-389	IX isolation	2	I	1
CH-390	RSSH to IX isolation	2	I	1
CH-391	IX to SWMS isolation	2	I	1
CH-392	IX isolation	2	I	1
CH-393	Regenerative HX vent	2	I	1
CH-394	IX bypass	2	I	1
CH-395	IX isolation	2	I	1
CH-396	SCS check	2	I	1
CH-397	SCS isolation	2	I	1
CH-398	IX isolation	2	I	1
CH-399	RSSH to IX isolation	2	I	1
CH-400	IX to SWMS isolation	2	I	1
CH-401	IX vent to GWMS	2	I	1
CH-402	IX resin fill isolation	2	I	1
CH-403	IX check	2	I	1
CH-404	IX isolation	2	I	1
CH-407, 408	IX D/P isolation	2	I	1
CH-414	Letdown strainer bypass	2	I	1
CH-415	IX isolation	2	I	1
CH-418	Letdown to VCT isolation	2	I	1
CH-419	Letdown strainer to SWMS	2	I	1
CH-420	IX effluent sample	2	I	1
CH-425	Charging line pressure indicator isolation	2	I	1
CH-426	Letdown sample isolation	2	I	1
CH-427, 428	Charging line flow indicator isolation	2	I	1
CH-431	Auxiliary spray check	1	I	1

TABLE 3.2-2 (Cont'd)
(Sheet 6 of 15)
SAFETY CLASS 1, 2 & 3 VALVES

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
CH-433	Charging line check	1	I	1
CH-434	Charging bypass line	1	I	1
CH-435	Charging line bypass check	2	I	1
CH-447	Auxiliary spray check	1	I	1
CH-448	Charging line check	1	I	1
CH-450	RDH to EDT check	3	I	1
CH-459	EDT line to GWMS pressure indicator isolation	3	I	1
CH-460, 461	EDT level indicator	3	I	1
CH-462	EDT drain isolation	3	I	1
CH-464	EDT to RDP check	3	I	1
CH-465, 466	RDP suction isolation	3	I	1
CH-468, 469	RDP discharge pressure indicator isolation	3	I	1
CH-470, 471	RDP discharge check	3	I	1
CH-472, 473	RDP discharge isolation	3	I	1
CH-474	Reactor drain filter bypass	3	I	1
CH-475	RDP discharge to RDH	3	I	1
CH-476	Reactor drain filter D/P	3	I	1
CH-477, 478	Reactor drain filter	3	I	1
CH-479	Reactor drain filter D/P	3	I	1
CH-480	IDH to EDT check	3	I	1
CH-485	Pre-holdup IX to RSSH	3	I	1
CH-486	Pre-holdup IX to IDH	3	I	1
CH-488	Pre-holdup IX D/P	3	I	1
CH-489	Pre-holdup strainer to SWMS isolation	3	I	1
CH-490	Pre-holdup IX isolation	3	I	1
CH-491	Pre-holdup IX strainer	3	I	1
CH-492	Pre-holdup IX D/P	3	I	1
CH-493	Pre-holdup IX effluent sample isolation	3	I	1
CH-494	RSSH and RDP to RDH Check	2	I	1
CH-495	Pre-holdup IX to BAST	3	I	1
CH-496	Pre-holdup IX to stripper	3	I	1
CH-500	VCT bypass valve	2	I	1
CH-501, 504	VCT discharge isolation	2	I	1
CH-505, 506	RCP CBO contain. isol.	2	I	1
CH-507	RCP bleedoff relief isol.	2	I	1
CH-510	BAST recirculation control	3	I	1

TABLE 3.2-2 (Cont'd)
(Sheet 7 of 15)
SAFETY CLASS 1, 2 & 3 VALVES

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
CH-512	VCT makeup isolation	3	I	1
CH-513	VCT to GWMS isolation	2	I	1
CH-514	BAMP to charging pump	3	I	1
CH-515	Letdown isolation	1	I	1
CH-516	Letdown backup isolation	1	I	1
CH-520	Ion exchanger bypass	2	I	1
CH-523	Letdown isolation	2	I	1
CH-524	Charging line isolation	2	I	1
CH-527	RMW to charging pump	3	I	1
CH-530, 532	BAST discharge isolation	2	I	1
CH-534	BAST to charging pump	2	I	1
CH-536	BAST to charging pump	3	I	1
CH-560	RDT suction isolation	2	I	1
CH-561	RDT isolation	2	I	1
CH-562	RDH to EDT isolation	3	I	1
CH-563	EDT to RDP isolation	3	I	1
CH-564	EDT to GWMS isolation	3	I	1
CH-565	Pre-holdup IX bypass	3	I	1
CH-566	Gas stripper bypass to EDT	3	I	1
CH-567	Gas stripper to VCT control	3	I	1
CH-580	RMWS to RDT isolation	2	I	1
CH-590, 591	Charging flow control	2	I	1
CH-600, 601, 602, 603	Letdown orifice isolation	2	I	1
CH-612, 613	Seal injection line vent	2	I	1
CH-614	Seal injection vent	3	I	1
CH-639	Chemical addition check	2	I	1
CH-645	VCT gas supply isolation	2	I	1
CH-646	RCP bleedoff check	2	I	1
CH-647	BAST recirc check	2	I	1
CH-648	BAST recirc sample	3	I	1
CH-649	BABE isolation	3	I	1
CH-653	F-210Y isolation	3	I	1
CH-654	MSH to gas stripper	3	I	1
CH-655	Pre-holdup IX to radiation monitor	3	I	1
CH-656	Gas stripper to HT	3	I	1
CH-657	EDT relief to misc radioactive sump	3	I	1
CH-660	Gas stripper inlet	3	I	1
CH-663	Reactor drain filter vent	3	I	1

TABLE 3.2-2 (Cont'd)**(Sheet 8 of 15)****SAFETY CLASS 1, 2 & 3 VALVES**

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
CH-665	RDP discharge sample	3	I	1
CH-668	BAM line to VCT check	3	I	1
CH-686	Holdup pump bypass to reactor drain filter	3	I	1
CH-700	Hydrostatic test connection	2	I	1
CH-701	Charging pump suction	2	I	1
CH-702	Charging suction pressure	2	I	1
CH-703, 704	Charging pump bypass	2	I	1
CH-705, 706	Charging pump discharge	2	I	1
CH-710	Hydrostatic test connection	2	I	1
CH-712	Charging suction pressure	2	I	1
CH-713	Charging pump suction	2	I	1
CH-714, 715	Charging pump bypass	2	I	1
CH-716	Charging miniflow HX vent	3	I	1
CH-717	Charging recirc relief	2	I	1
CH-719	Charging pump discharge	2	I	1
CH-721, 722	Letdown to pre-holdup IX	3	I	1
CH-723	Reactor drain line sample	3	I	1
CH-724	Pre-holdup IX isolation	3	I	1
CH-725	Pre-holdup IX check	3	I	1
CH-726	Pre-holdup IX resin fill	3	I	1
CH-727	Pre-holdup IX D/P	3	I	1
CH-728	Pre-holdup IX vent	3	I	1
CH-730	Pre-holdup IX to SWMS	3	I	1
CH-747	Charging line check	2	I	1
CH-750	Charging flow control iso.	2	I	1
CH-751	Regen. HX charging isolation	2	I	1
CH-753	BAMP to PCPS	3	I	1
CH-754	Charging flow control iso.	2	I	1
CH-760, 761	Charging recirculation check	2	I	1
CH-763	Charging pump discharge	2	I	1
CH-764, 766	Charging flow control iso.	2	I	1
CH-768	Chemical addition line	2	I	1
CH-769, 770	Charging miniflow HX iso.	2	I	1
CH-787	Seal injection check	1	I	1
CH-789, 800	Seal injection flow	2	I	1
CH-802	Seal injection check	1	I	1
CH-804, 805	Seal injection flow	2	I	1
CH-807	Seal injection check	1	I	1
CH-809, 810	Seal injection flow	2	I	1

TABLE 3.2-2 (Cont'd)**(Sheet 9 of 15)****SAFETY CLASS 1, 2 & 3 VALVES**

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
CH-812	Seal injection check	1	I	1
CH-814, 815	Seal injection flow indicator	2	I	1
CH-816, 818, 819, 821	Seal injection filter	2	I	1
CH-822, 823	Seal injection to DRDH	2	I	1
CH-825, 826	Seal injection filter D/P	2	I	1
CH-830	Nitrogen supply to EDT	3	I	1
CH-831	Nitrogen supply pressure control	3	I	1
CH-835	Seal injection check	2	I	1
CH-839	Seal injection isolation	2	I	1
CH-844, 845	Seal injection filter vent	2	I	1
CH-858	RSSH line to EDT check	3	I	1
CH-861	RSSH to EDT isolation	3	I	1
CH-862	RMWT supply to RDT	3	I	1
CH-865	Seal injection relief	2	I	1
CH-866, 867, 868, 869	Seal injection check	1	I	1
Pool Cooling and Purification System (PCPS) (1)				
PC-200, 210	Cooling HX inlet pressure	3	I	1
PC-201, 293	Cooling HX cross-connect	3	I	1
PC-202, 203	Cooling pump suction isolation	3	I	1
PC-204, 205	Cooling pump discharge pressure	3	I	1
PC-206, 207	Cooling pump discharge check	3	I	1
PC-208, 209	Cooling pump disch. isolation	3	I	1
PC-211, 212	Cooling HX inlet isolation	3	I	1
PC-213, 214	Cooling HX outlet isolation	3	I	1
PC-249	IRWST return line isolation	3	I	1
PC-257, 258	Refueling pool disch. isolation	2	I	1
PC-291, 292	Refueling pool inlet isolation	2	I	1
PC-300, 301, 302, 303	Cooling flow indication iso.	3	I	1
Safety Depressurization System (SDS)				
RC-406, 407, 408, 409	Rapid depressurization	1	I	1
RC-410, 411, 412, 413	Pressurizer vent	1	I	1
RC-414, 415, 416, 417	Reactor vessel vent	2	I	1
RC-418	RCGVS vent to RDT	2	I	1
RC-419	RCGVS vent to IRWST	2	I	1
RC-263, 264	RD pressure indication	2	I	1
RC-267	RCGVS pressure indication	2	I	1

TABLE 3.2-2 (Cont'd)**(Sheet 10 of 15)****SAFETY CLASS 1, 2 & 3 VALVES**

<u>Component Identification</u>	<u>Location/ Description</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
Safety Injection System (SIS) (1)				
SI-100, 101	IRWST return check valve	2	I	1
SI-102, 103	IRWST isolation valve test	2	I	1
SI-104, 105	CSP suction isolation	2	I	1
SI-106, 107	SCS suction isolation	2	I	1
SI-108, 109	Pressure gauge connection	2	I	1
SI-110, 111	SCS/CS pump suction cross-connect	2	I	1
SI-113, 123, 133, 143	Safety inj. containment check	1	I	1
SI-115, 125, 135, 145	SI flow indication isolation	2	I	1
SI-116, 126, 136, 146	SI flow indication isolation	2	I	1
SI-117, 127, 137, 147	SIT pressure indication iso.	2	I	1
SI-119, 129, 139, 149	SIT pressure indication iso.	2	I	1
SI-120, 121	SCS suction BAMP isolation	2	I	1
SI-130, 131	Safety injection pump isolation (Pumps 3 and 4)	2	I	1
SI-157, 158	Containment spray pump check	2	I	1
SI-161, 193	CSS IRWST recirc line relief	2	I	1
SI-162, 192	IRWST recirculation line refueling pool relief	2	I	1
SI-164, 165	Containment spray check	2	I	1
SI-166	SI HL 2 to EDT relief	2	I	1
SI-168, 178	SCS check valve	2	I	1
SI-169	SCS 2 suction thermal relief	1	I	1
SI-170, 180	SDCHX vent valve	2	I	1
SI-172, 182	SDCHX drain valve	2	I	1
SI-174, 175, 176, 177	CSP discharge flow indication	2	I	1
SI-179, 189	SCS suction line relief valve	2	I	1
SI-187, 188	SCS IRWST recirc line relief	2	I	1
SI-191, 194	CS HX to EDT relief valve	2	I	1
SI-196, 197	Pressure gauge connection	2	I	1
SI-207, 208	IRWST isolation valve test	2	I	1
SI-210, 220, 230, 240	SIT fill and drain isolation	2	I	1
SI-211, 221, 231, 241	SIT relief valve	2	I	1
SI-212, 222, 232, 242	SIT level indication isolation	2	I	1
SI-213, 223, 233, 243	SIT level indication isolation	2	I	1
SI-214, 224, 234, 244	SIT local sample isolation	2	I	1
SI-215, 225, 235, 245	SIT check valve	1	I	1
SI-216, 226, 236, 246	Injection line pressure iso.	2	I	1

TABLE 3.2-2 (Cont'd)**(Sheet 11 of 15)****SAFETY CLASS 1, 2 & 3 VALVES**

<u>Component Identification</u>	<u>Location/ Description</u>	<u>Safety Class</u>	<u>Seismic Category</u>	<u>Quality Class</u>
SI-217, 227, 237, 247	Safety injection line check	1	I	1
SI-218, 219, 254, 255	SI pump orifice bypass valve	2	I	1
SI-228, 238, 248, 258	SIT level indication isolation	2	I	1
SI-229, 239, 249, 259	SIT level indication isolation	2	I	1
SI-250, 251	CS miniflow sample isolation	2	I	1
SI-252, 253	SCS miniflow sample isolation	2	I	1
SI-260, 264	SDCHX vent valve	2	I	1
SI-261, 267	CS pump miniflow isolation	2	I	1
SI-262, 266	SDCHX drain valve	2	I	1
SI-265, 269	SCS pump miniflow isolation	2	I	1
SI-285, 286	Safety injection to EDT relief	2	I	1
SI-287, 289	SCS IRWST recirc line relief	2	I	1
SI-290, 291	SIT fill line isolation	2	I	1
SI-292	EDT/SIT relief valve	2	I	1
SI-293	SIT isolation valve	2	I	1
SI-294	EDT/SIT local sample	2	I	1
SI-300, 301	CS/SCS IRWST recirc isolation	2	I	1
SI-302, 303	SI IRWST recirc line isolation	2	I	1
SI-304, 305, 308, 309	IRWST isolation valve	2	I	1
SI-310, 311	SDCHX flow control valve	2	I	1
SI-312, 313	SDCHX bypass flow control	2	I	1
SI-314, 315	SCS IRWST recirc flow control	2	I	1
SI-321, 331	Hot leg inj. loop isolation	2	I	1
SI-322, 332	Hot leg check valve leak iso.	2	I	1
SI-390, 391, 392, 393	Holdup volume tank spillway	2	I	1
SI-394, 395	Reactor cavity spillway	2	I	1
SI-402, 470	Safety injection pump isolation (Pumps 1 and 2)	2	I	1
SI-404, 405, 434, 446	SI pump discharge check	2	I	1
SI-408, 416, 433, 436	Pressure gauge isolation	2	I	1
SI-409, 417, 439, 449	SI pump discharge to EDT relief	2	I	1
SI-410, 411, 412, 413	SI pump miniflow isolation	2	I	1
SI-418, 419	Shutdown purification isolation	2	I	1
SI-420, 421	Shutdown purification isolation	2	I	1
SI-424, 426, 448, 451	SI pump bypass check valve	2	I	1
SI-427, 465	SI effluent sampling valve	2	I	1
SI-429, 445	SCS suction sample isolation	2	I	1
SI-430, 431	SCS/CSS discharge cross-connect	2	I	1
SI-435, 447, 476, 478	SI pump discharge isolation	2	I	1
SI-450, 458	IRWST recirculation line refueling pool isolation	2	I	1

TABLE 3.2-2 (Cont'd)

(Sheet 12 of 15)

SAFETY CLASS 1, 2 & 3 VALVES

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
SI-454, 455	IRWST recirculation line refueling pool isolation	2	I	1
SI-466, 467	SCS to EDT relief valve	2	I	1
SI-468	SI HL 1 to EDT relief valve	2	I	1
SI-469	SCS 1 suction thermal relief	1	I	1
SI-473	SIT fill line thermal relief	2	I	1
SI-474	RDT relief valve	2	I	1
SI-482, 483	CSP discharge pressure indic.	2	I	1
SI-484, 485	CSP discharge check valve	2	I	1
SI-488, 489	CSP discharge isolation	2	I	1
SI-490, 491, 492, 493	SCS flow indication isolation	2	I	1
SI-500, 501, 510, 511	CS header test connection	2	I	1
SI-506, 516	SI HL pressure indication	2	I	1
SI-522, 532	SI hot leg inj. check	2	I	1
SI-523, 533	SI HL containment check	2	I	1
SI-525, 526	SI HL 1 flow indication	2	I	1
SI-535, 536	SI HL 2 flow indication			
SI-550, 552, 553, 555	SI pump test isolation valve	2	I	1
SI-560, 561, 562, 563	CS miniflow HX vent valve	2	I	1
SI-564, 565, 566, 567	CS miniflow HX drain valve	2	I	1
SI-568, 569	SCS pump outlet check valve	2	I	1
SI-570, 571, 572, 573	SCS miniflow HX vent valve	2	I	1
SI-574, 575, 576, 577	SCS miniflow HX drain valve	2	I	1
SI-578, 579	SCS pump outlet isolation	2	I	1
SI-580, 581, 582, 583	CS heat exchanger vent valve	2	I	1
SI-584, 585, 586, 587	CS heat exchanger drain valve	2	I	1
SI-588, 589	IRWST recirc line BABE isolation	2	I	1
SI-600, 601	SCS train isolation valve	2	I	1
SI-602, 603	SI low flow control valve	2	I	1
SI-604, 609	SI hot leg inj. isolation	2	I	1
SI-605, 606, 607, 608	SIT atmospheric vent isolation	2	I	1
SI-611, 621, 631, 641	SIT fill and drain isolation	2	I	1
SI-612, 622, 632, 642	Nitrogen pressure control	2	I	1
SI-613, 623, 633, 643	SIT atmospheric vent isolation	2	I	1
SI-614, 624, 634, 644	SIT discharge isolation valve	1	I	1
SI-616, 626, 636, 646	Injection line isolation	1	I	1
SI-618, 628, 638, 648	Check valve leakage isolation	2	I	1
SI-619, 629, 639, 649	Nitrogen pressure control	2	I	1
SI-651, 652, 653, 654	SCS suction line isolation	2	I	1
SI-655, 656	SCS suction line isolation	2	I	1

TABLE 3.2-2 (Cont'd)**(Sheet 13 of 15)****SAFETY CLASS 1, 2 & 3 VALVES**

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
SI-657, 658	CSS IRWST recirc flow control	2	I	1
SI-661	RDT isolation valve	2	I	1
SI-670	IRWST return isolation	2	I	1
SI-671, 672	Containment spray header iso.	2	I	1
SI-682	SIT fill line isolation	2	I	1
SI-686, 696	CS HX to IRWST isolation	2	I	1
SI-687, 695	CS header block valve	2	I	1
SI-688, 693	SCS IRWST recirc isolation	2	I	1
SI-690, 691	SCS train warm-up flow control	2	I	1
SI-700, 701, 702, 703	IRWST level indication	2	I	1
SI-704, 705, 706, 707	HVT level indication	2	I	1
SI-708, 709, 710, 711	Reactor cavity level indication	2	I	1
SI-712, 713, 714, 715	IRWST vacuum breakers	2	I	1
SI-716, 717, 718, 719	IRWST relief valves	2	I	1
SI-720	IRWST return header isolation	2	I	1
Emergency Feedwater System (EFW) (1)				
EF-100, 101	Steam-Driven Pump Isolation	2	I	1
EF-102, 103	Motor-Driven Pump Isolation	2	I	1
EF-104, 105	Steam-Driven Pump Flow Control	3	I	1
EF-106, 107	Motor-Driven Pump Flow Control	3	I	1
EF-108, 109	Steam-Driven Pump Turbine			
	Steam Supply Isolation	2	I	1
EF-110, 111	Steam Supply Drain Isolation	2	I	1
EF-112, 113	Steam Supply Bypass Isolation	2	I	1
EF-200, 201, 202, 203	SG Isolation Check Valve	2	I	1
EF-204, 205, 206, 207	EFW Pump Discharge Check	3	I	1
EF-208, 209, 210, 211	EFW Pump Suction Isolation	3	I	1
EF-212, 213	EFWST Crossover Isolation	3	I	1
EF-214, 215	Non-Safety Condensate Source			
	Isolation Check Valve	3	I	1
EF-216, 217	EFWST Drain Isolation Valve	3	I	1
EF-220, 221, 222, 223	EFW Pump Minimum Flow Isolation	3	I	1
EF-224, 225, 226, 227	Full Flow Test Bypass Isolation	3	I	1
EF-228, 229, 230, 231	Full Flow Test Flow Control	3	I	1
EF-232, 233, 234, 235	Full Flow Test Bypass Isolation	3	I	1
EF-236, 237	Steam-Driven Pump Turbine Bearing			
	Oil Cooler Return Isolation	3	I	1

TABLE 3.2-2 (Cont'd)**(Sheet 14 of 15)****SAFETY CLASS 1, 2 & 3 VALVES**

Component Identification	Location/ Description	Safety Class	Seismic Category	Quality Class
EF-238, 239	Steam Supply Maintenance Iso.	2	I	1
EF-240, 241, 244, 245, 246, 247, 248, 249	Steam Supply Drain Isolation	2	I	1
EF-250, 251, 252, 253, 254, 255	Steam Exhaust Drain Isolation	3	I	1
EF-256, 257, 258, 259, 260, 261, 262, 263	Flow Indicator Isolation	3	I	1
EF-264, 265, 266, 267, 268, 269, 270, 271	Pressure Indicator Isolation	3	I	1
EF-272, 273, 274, 275, 276, 277, 278, 279	Flow Indicator Isolation	3	I	1
EF-280, 281, 282, 283	Pressure Test Isolation	3	I	1
EF-284, 285, 286, 287	Level Indication Isolation	3	I	1
EF-288, 289, 290, 291	EFW Pump Discharge Crossover	3	I	1
EF-292, 293	Pressure Indication Isolation	2	I	1
EF-294, 295	Pressure Test Isolation	2	I	1
EF-296, 297	Pressure Test Isolation	3	I	1
EF-298, 299	Level Switch Isolation	2	I	1
EF-300, 301	Level Switch Isolation	3	I	1
EF-310, 311	EFWST Cleanup Isolation	3	I	1
EF-316, 317, 318, 319, 320, 321, 322, 323, 324, 325, 326, 327	Steam Supply Drain Isolation	2	I	1
EF-328, 329	Turbine Case Drain Isolation	3	I	1
EF-330, 331	Steam Supply Bypass Maintenance	2	I	1
EF-334, 335, 336, 337	Level Indication Isolation	3	I	1
EF-338, 339, 340, 341	EFW Pump Discharge Maintenance	3	I	1

TABLE 3.2-2 (Cont'd)

(Sheet 15 of 15)

SAFETY CLASS 1, 2 & 3 VALVES

NOTE: (1) All containment isolation valves and their operators, including manual valves, check valves, and relief valves which also serve as isolation valves, are subject to the pertinent requirements of the Quality Assurance Program as given in Chapter 17. I

TABLE 3.2-3RELATIONSHIP OF SAFETY CLASS TO CODE CLASS

<u>Safety Class</u>	<u>Code Class (ASME Section III)</u>	
SC-1	1	I
SC-2 for reactor containment components	2	
SC-2 for fluid system components	2	D
SC-3 for core support structures	CS	
SC-3 (otherwise)	3	
NNS	Industry Standards	

TABLE 3.2-4

SUMMARY OF CRITERIA - STRUCTURES

Structures	Loading								Remarks
	Normal Wind	Dead and Equipment	Live	Containment Accident Pressure	Seismic OBE	SSE	Tornado Wind	Missile	Including Any Environmental Requirements
Containment	-	X	X	X	X	X	-	-	Thermal Stresses; Equipment Missile Protected
Containment Interior Concrete	-	X	X	-	X	X	-	-	Differential Accident Pressure; Pipe Rupture Loads; Thermal Stresses; Equipment Missile Protected
Shield Building Including Foundation	X	X	X	-	X	X	X	X	Tornado Pressure Differential; Soil and Water Pressure
Containment Penetrations	-	X	X	X	X	X	-	-	Pipe Whipping
Containment Structural Steel	-	X	X	-	X	X	-	-	Thermal Stresses
Underground Cable Systems	X	X	X	-	X	X	-	-	
Station Service Water Pipe	X	X	X	-	X	X	X	X	Soil and Water Pressures or Buried Portion Hydraulic Pressures; Moving Equipment Loads
Station Service Water Structures Including Pump Structure	X	X	X	-	X	X	X	X	Soil and Water Pressures; Seismic Analysis
Intake & Discharge Structure									
Dam									
Spillway Control Structure									
Auxiliary Building Including Control Building	X	X	X	-	X	X	X	X	Soil and Water Pressures on Substructure; Tornado Pressure Drop;
Diesel Building									Thermal Stresses and Cask Drop;
Fuel Pool									Pipe and Pipe Rupture Loads;
Main Steam and Feedwater Enclosures									
Fuel Storage Racks	-	X	X	-	X	X	-	-	Thermal Stresses
Station Test	X	X	X	-	X	X	-	-	

3.3 WIND AND TORNADO LOADINGS

All Seismic Category I structures, except those not exposed to wind, are designed for wind and tornado loadings.

3.3.1 WIND LOADINGS

The design for wind loading is in accordance with ANSI A58.1, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures" (Reference 1), ASCE Paper 3269, "Wind Forces on Structures" (Reference 2), and ASCE Paper 4933, "Wind Loads on Dome-Cylinder and Dome-Cone Shapes" (Reference 3).

3.3.1.1 Design Wind Velocity

A design wind velocity of 130 mph, at a height of 30 feet above nominal ground elevation is used as the fastest mile of wind for a 100 year recurrence period.

Vertical wind velocity profiles and associated effective pressures are calculated in accordance with Section 6 of Reference 1 utilizing an Importance Factor, I, of 1.11 and Exposure C.

Gust response factors are dependent on height above grade level and are in accordance with Table 8 of Reference 1 for Exposure C.

3.3.1.2 Determination of Applied Forces

Based on structure geometry and physical configuration, the effective pressure distribution is transformed into applied equivalent static building forces utilizing appropriate shape coefficients given in References 1 and 3.

Wind pressure distribution curves for the containment shield building are shown in Figure 3.3-1. The maximum height of the shield building above grade is approximately 173 feet 3 inches.

3.3.2 TORNADO LOADINGS

All Seismic Category I structures, except those structures not exposed to wind, are designed for tornado loadings.

3.3.2.1 Applicable Design Parameters

Tornado effects are in accordance with Interim Regulatory Guide 1.76 (Reference 4). The following parameters are applicable to the design basis tornado.

Maximum wind speed:	330 mph
Rotational speed:	260 mph
Translational velocity:	70 mph
Radius:	150 feet
Maximum pressure differential:	2.4 psid
Rate of pressure drop:	1.7 psi/second
Missile Spectra:	See Table 3.5-2

3.3.2.2   Determination of Forces on Structures

The forces on Seismic Category I structures due to tornado wind loadings are obtained using methods outlined in Section 3.3.1.2, with a wind velocity of 330 mph (vector sum of all component velocities - assumed constant with height). Velocity profiles are determined as outlined in Section 3.3.1.1. Effective pressure distribution loads are transformed into equivalent static building forces as outlined in Section 3.3.1.2. In determining tornado wind loadings, both the importance factor and gust factors are taken as unity.

Tornado loadings include tornado wind pressure, internal pressure due to tornado created atmospheric pressure drop, and forces generated due to the impact of credible tornado missiles. These loadings are combined with other loads as described in Section 3.8.

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

Adjacent structures will not be permitted to affect or degrade the capability of Seismic Category I structures to perform their intended safety functions as a result of tornado loadings. This is accomplished by one of the following methods:

- A. Designing the adjacent structure to Seismic Category I tornado loadings.
- B. Investigating the effect of adjacent structural failure on Seismic Category I structures to determine that no impairment of function results.
- C. Designing a structural barrier to protect Seismic Category I structures from adjacent structural failure.

REFERENCES FOR SECTION 3.3

1. "Minimum Design Loads for Buildings and Other Structures," ANSI A58.1-1982. D
2. "Wind Forces on Structures," ASCE Paper No. 3269, Transactions, ASCE, Vol. 126, Part II, 1961, p. 1124.
3. "Wind Loads on Dome-Cylinder and Dome-Cone Shapes," ASCE Paper No. 4933, Journal of the Structural Division - Proceedings of the American Society of Civil Engineers, Vol. 92, No. ST5, October 1966.
4. Safety Evaluation by the Office of Nuclear Reactor Regulation of Recommended Modification to the R.G. 1.76 Tornado Design Basis for the ALWR, attached to a March 25, 1988 NRC letter to the ALWR Utility Steering Committee. I

3.4   WATER LEVEL (FLOOD) DESIGN

All Seismic Category I structures, components and equipment are designed for applicable loadings caused by postulated floods. Section 2.4 of the site-specific SAR describes, in detail, the relationship of the site-specific flood levels to safety-related buildings and facilities.

3.4.1   FLOOD ELEVATIONS

The elevation level for floods at the reactor site is determined in accordance with Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," and ANSI/ANS 2.8-1983, "Determining Design Basis Flooding at Power Reactor Sites." The design basis level for the System 80+ Standard Design is limited to 1 foot below plant finished yard grade as the minimum flood level value. The maximum flood level value is site-specific and protection measures for that flood level are described in Section 2.4 of the site-specific SAR.

3.4.2   PHENOMENA CONSIDERED IN DESIGN LOAD CALCULATION

All safety-related structures of the reactor building complex are designed to withstand the static and dynamic forces of the plant flood level. Other safety-related structures or systems essential for plant operation are designed for the site-related flood level as described in Section 2.4 of the site-specific SAR.

3.4.3   FLOOD FORCE APPLICATION

The design flood is used in determining the applicable water level for design of all Seismic Category I structures in accordance with the load combinations discussed in Section 3.8.4. The forces acting on those structures are determined on the basis of full external hydrostatic pressure corresponding to that flood level. All Seismic Category I structures will be in a stable condition due to both moment and uplift forces resulting from the proper load combinations, including design basis flood levels.

3.4.4   FLOOD PROTECTION

3.4.4.1   Flood Protection Measures for Seismic Category I Structures

The flood protection measures for Seismic Category I structures, systems and components are designed in accordance with Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants." The following structures and systems in the reactor complex area are designed for flood level protection:

- Reactor Building
- Diesel Generator Buildings
- Control Building
- Fuel Building
- Auxiliary Building

These safety-related structures are designed to maintain a dry environment during all floods by incorporating the following safeguards into their construction:

- A. No exterior access openings will be lower than 1 foot above plant grade elevation.
- B. The finished yard grade adjacent to the safety-related structures will be maintained at least 1 foot below the ground floor elevation.
- C. Waterstops are used in all horizontal and vertical construction joints in all exterior walls up to flood level elevation.
- D. Water seals are provided for all penetrations in exterior walls up to flood level elevation.
- E. Waterproofing of walls subject to flooding is provided.

For other safety related structures where flood protection measures are required (e.g. pumping systems, stoplogs, watertight doors, dikes, retaining walls and drainage systems) the design of means for providing such protection will be described in Section 2.4 of the site-specific SAR.

Redundant equipment is separated and compartmentalized so that a single flooding event does not affect redundant safety systems. Equipment such as the auxiliary shutdown panels are elevated off the floor so that flooding events will not affect these important pieces of equipment.

3.4.4.2   Permanent Dewatering System

A permanent safety grade dewatering system will be installed to maintain the groundwater table at or below elevation 40 + 0. For sites where the groundwater table is at or below the foundation, the system is not required.

3.4.5 ANALYTICAL AND TEST PROCEDURES

A description of the methods and test procedures by which static and dynamic effects of the design basis flood conditions or design basis groundwater conditions are applied is detailed in Section 2.4 of the site-specific SAR.

D

3.5 MISSILE PROTECTION

The missile protection design for Seismic Category I structures, systems and components is described in this section.

Missile protection or redundancy is provided for Seismic Category I equipment and components such that internal and external missiles will not cause the release of significant amounts of radioactivity or prevent the safe and orderly shutdown of the reactor.

The protection of essential structures, systems and components will be accomplished by one or more of the following:

- A. Minimizing the sources of missiles by equipment design features that prevent missile generation.
- B. Orientation or physical separation of potential missile sources away from safety-related equipment and components.
- C. Containment of potential missiles through the use of protective shields and barriers near the source.
- D. The hardening of safety-related equipment and components to withstand missile impact, where such impacts cannot be reasonably avoided by the methods above.

3.5.1 MISSILE SELECTION AND DESCRIPTION

Potential missiles are identified and characterized by type and source and their probability of occurrence, retention and impact. For equipment with energy sources capable of creating a missile, the selection is based on the application of a single-failure criterion to the retention features of the component. Where sufficient retention redundancy is provided in the event of a failure, no missile is postulated.

Internally generated missiles can be generated potentially from two types of equipment: rotating components and pressurized components. Rotating components include turbine wheels, fans, auxiliary pumps and their associated motors. Pressurized components include valves, heat exchangers, vessels and their associated components.

The types of missiles considered and/or not considered in the design of Seismic Category I structures, systems, and components are discussed in the following sections:

- A. Internally Generated Missiles (Outside Containment), described in Section 3.5.1.1.
- B. Containment Internal Missiles, defined in Table 3.5-1 and Section 3.5.1.2.
- C. Turbine Missiles, described in Section 3.5.1.3.
- D. Natural Phenomena (Tornado) Missiles, described in Section 3.5.1.4.
- E. Site Proximity Missiles (Except Aircraft), described in Section 3.5.1.5.
- F. Aircraft Hazards, described in Section 3.5.1.6.

3.5.1.1 Internally Generated Missiles (Outside Containment)

Internally generated missiles (outside containment) from rotating and pressurized components are not considered credible for the reasons discussed below.

The redundant safety systems outside of containment are physically separated such that no single gravitational or other type missile can impact both systems.

3.5.1.1.1 Auxiliary Pumps and Motors

There are no postulated missiles originating from auxiliary pumps and associated motors outside containment for the following reasons:

- A. The pump motors are induction type which have relatively slow running speeds and are not prone to overspeed. The motors are all pretested at full running speed by the motor vendor prior to installation.
- B. In addition to the low likelihood of missiles due to motor overspeed as discussed in A. above, the motor stator would tend to serve as a natural container of rotor missiles if there were to be any.
- C. All pumps normally have relatively low suction pressures and, therefore, would not tend to be driven to overspeed due to a pipe break in the discharge line. In addition, the induction motor would tend to act as a brake to prevent pump overspeed.

- D. Industry pump designs are such that (and service history shows) no occurrences of impeller pieces penetrating pump casings.

3.5.1.1.2 Valves

There are no missiles postulated from valves for the following reasons:

- A. All valve stems are provided with a backseat or shoulder larger than the valve bonnet opening.
- B. Motor operated and manual valve stems are restrained by stem threads.
- C. Operators on motor, hydraulic and pneumatic operated valves prevent stem ejection.
- D. Pneumatic operated diaphragms and safety valve stems are restrained by spring force.
- E. All valve bonnets are either pressure sealed, threaded or bolted such that there is redundant retention for prevention of missile generation.

3.5.1.1.3 Pressure Vessels

All pressurized vessels are considered moderate energy (275 psig) or less and are designed and constructed to the standards of the ASME Code. In addition to the ASME Code examination and testing requirements, all vessels will receive periodic in-service inspections. Where appropriate, these components are provided with pressure relief devices to ensure that no pressure buildup will exceed material design limits.

On this basis, moderate energy pressure vessels are not considered credible missile sources.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Table 3.5-1 lists postulated missiles from equipment inside containment, and summarizes their characteristics. Included are major pretensioned studs and nuts, instruments, and the CEDM missile. Other items which were considered and specifically excluded because of redundant retention features are valve stems, valve bonnets and pressurized cover plates.

3.5.1.3 Turbine Missiles

The probability of turbine missile generation and adverse impact effects on Seismic Category I systems and components is assured to be less than $1.0E-3$ events per turbine-year by a combination of the following measures:

- A. Reliable turbine overspeed protection provisions (see Section 10.2.2 for details).
- B. Adequate assurance of turbine disc integrity by design and inspection (see Sections 10.2.2 and 10.2.4 for details).
- C. Placement and orientation of the turbine generator (described below).
- D. Consideration of the protection provided by plant structures not explicitly designed as barriers that may limit missile penetrating capabilities to less than the capability of Seismic Category I structures.

The turbine generator placement and orientation for the System 80+ Standard Design, and the corresponding low-trajectory missile strike zones, are illustrated in Figure 1.2-1. The placement and orientation of the turbine generator provides adequate protection against low trajectory turbine missiles by excluding safety-related structures, systems, and components from the low trajectory turbine missile strike zones in accordance with the guidelines of Regulatory Guide 1.115.

Critical structures (i.e., those housing safety-related equipment) and exterior equipment are located in line with, or within close proximity to, the longitudinal axis of the turbines. This makes the potential for turbine-generated missiles to strike these targets negligibly small.

The System 80+ design follows the guidelines of Regulatory Guide 1.115 by placing and orienting the turbine such that all safety-related structures, systems, and components are excluded from the low trajectory turbine missile strike zones or if site characteristics make this impossible, safety-related targets will be placed and shielded such that the combined strike and damage probability for the safety-related targets in these zones is less than 10^{-3} per turbine failure.

3.5.1.4 Missiles Generated by Natural Phenomena

Tornado-generated missiles are the limiting natural hazard and, as such, are a part of the design basis for Seismic Category I structures and components. Table 3.2-4 lists those structures,

shields and barriers that will be designed for tornado missile effects. Tornado-generated missiles considered in the design are given in Table 3.5-2. D
I

3.5.1.5 Missiles Generated by Events Near the Site

Justification will be provided in the site-specific SAR. D

3.5.1.6 Aircraft Hazards

Justification will be provided in the site-specific SAR. Also refer to Section 2.2.1. I

3.5.2 STRUCTURES, SYSTEMS, AND COMPONENTS TO BE PROTECTED FROM EXTERNALLY GENERATED MISSILES

Tornado missiles are the design basis missiles from external sources. All safety related systems, equipment and components required to safely shut the reactor down and maintain it in a safe condition are housed in Category I structures designed as tornado resistant (see Section 3.5.1.4) and as such are considered to be adequately protected.

3.5.3 BARRIER DESIGN PROCEDURES

Missile barriers, whether steel or concrete, are designed with sufficient strength and thickness to stop postulated missiles and to prevent overall damage to Seismic Category I structures. The procedures by which structures and barriers are designed to perform this function are presented in this section.

3.5.3.1 Local Damage Prediction

The prediction of local damage in the immediate vicinity of an impacted area depends on the basic material of construction of the barrier itself (i.e. either concrete or steel). Corresponding procedures are discussed separately below. D

3.5.3.1.1 Concrete Structures and Barriers

Local damage prediction for concrete structures includes the estimation of the depth of missile penetration and an assessment of whether secondary missiles might be generated by spalling. Generally, the Modified Petry Formula or the Modified NDRC Formula (References 2 and 3) is used to estimate missile penetration with appropriate constants taken from available test data. To insure that no secondary missiles (due to spalling) are generated, a minimum barrier thickness of 3 times the penetration depth is provided. Depending on certain missile characteristics, additional penetration formulas may be employed as justified by full scale impact tests (References 3 and 4).

3.5.3.1.2 Steel Structures and Barriers

The Stanford equation (Reference 5) is used as the basis for the design and analysis of steel structures and barriers.

3.5.3.2 Overall Damage Prediction

The overall response of a structure or barrier to missile impact depends largely on the location of impact (e.g. near mid-span or near a support), the dynamic and deformation properties of the barrier and the missile, and the kinetic energy of the missile itself.

Depending on the deformation characteristics of both the barrier and the missile, an impact force time history can be developed using either work-kinetic energy principles or conservation of momentum. The structural response to this impulse loading, in conjunction with other appropriate design loads, is evaluated by the procedures given in References 3 and 6.

3.5.4 INTERFACE REQUIREMENTS

Protection for all Seismic Category I structures, systems and components shall be provided by one of the following:

- A. For systems and parts of systems located inside the containment (RCS and connected systems, Engineered Safety Feature systems), appropriate missile barrier design procedures are used to ensure that the impact of any potential missile will not lead to a loss-of-coolant-accident or preclude the systems from carrying out their specified safety functions.
- B. For systems and equipment outside containment, appropriate design procedures (e.g., proper turbine orientation, natural separation, or missile barriers) are used to ensure that the impact of any potential missile does not prevent the system or equipment from carrying out its specified safety function.
- C. For all systems and equipment, appropriate design procedures are used to ensure that the impact of any potential missile does not prevent the conduct of a safe plant shutdown, or prevent the plant from remaining in a safe shutdown condition.
- D. Safety-related instrumentation and control equipment are protected from potential missile sources. The 1E and associated cabling and sensing lines are also protected from potential missile sources.

REFERENCES FOR SECTION 3.5

1. "Plant Design Against Missiles," ANSI/ANS-58.1. (DRAFT - Formerly ANSI N177-1974.)
2. A. Amirikan, "Design of Protective Structures," Report No. NT-3726, Bureau of Yards and Docks, Dept. of the Navy, August 1950.
3. "Structural Analysis and Design of Nuclear Power Plant Facilities," Manual No. 58, Chapter 6, American Society of Civil Engineers, 1980.
4. Stephenson, A. E., "Full Scale Tornado-Missile Impact Tests," EPRI NP-440, July 1977, Prepared for the Electric Power Research Institute by Sandia National Laboratories.
5. Cottrell, W. B. and A. W. Savolainen, "U.S. Reactor Containment Technology," ORNL-NSIC-5, Vol. 1, Chapter 6, Oak Ridge National Laboratory.
6. Williamson, R. A. and R. R. Alvy, "Impact Effects of Fragments Striking Structural Elements," Holmes and Narver, Inc., Revised November, 1973.

D

TABLE 3.5-1

(Sheet 1 of 2)

KINETIC ENERGY OF POTENTIAL MISSILES⁽¹⁾

Item⁽²⁾	Initial Kinetic Energy (ft-lb)	Weight (lb)	Impact Section
1. Reactor Vessel			
Closure Head Nut	1,706	100	Annular Ring, OD = 10.125" ID = 6.9"
Closure Head Nut and Stud	5,226	577	Solid Circle, 6.75" Diameter
Control Rod Drive Assembly	57,600	1100	1.875" Diameter Solid Circle within a Concentric 7" Diameter by .109" Wall Shroud
2. Steam Generator			
Primary Manway Stud and Nut	71	4.25	Solid Circle, 1.5" Diameter
Secondary Handhole Stud and Nut	7	1.15	Solid Circle, 0.75" Diameter
Secondary Manway Stud	7	3.36	Solid Circle, 1.25" Diameter
3. Pressurizer			
Safety Valve Flange Bolt	15	3.7	Solid Circle, 1.25" Diameter
Lower Temperature Element	288	3	Edge of Solid Disk 2.75" Diameter and 0.5" Thick
Manway Stud and Nut	71	4.25	Solid Circle, 1.5" Diameter
4. Reactor Coolant Pump and Piping			
Temperature Nozzle with RTD Assembly	1,095	8	Edge of Solid Disk 2.75" Diameter and 0.5" Thick

TABLE 3.5-1 (Cont'd)

(Sheet 2 of 2)

KINETIC ENERGY OF POTENTIAL MISSILES⁽¹⁾

<u>Item</u> ⁽²⁾	<u>Initial Kinetic Energy (ft-lb)</u>	<u>Weight (lb)</u>	<u>Impact Section</u>
Surge and Spray Piping Thermowells with RTD Assembly	277	3.75	Edge of Solid Disc 2.75" Diameter and 0.5" Thick
Reactor Coolant Pump Thermowell with RTD	1,095	8	Edge of Solid Disk 2.75" Diameter and 0.5" Thick

NOTES: (1) All dimensions, weights and kinetic energies are typical values.

(2) All materials are steel.

TABLE 3.5-2

DESIGN BASIS TORNADO MISSILES AND THEIR IMPACT VELOCITIES

<u>Missile Descriptions</u>	<u>Dimensions (m)</u>	<u>Weight (lbs)</u>	<u>Impact Area (in²)</u>	<u>Design Impact Velocity (ft/sec)</u>	
				<u>Horizontal</u>	<u>Vertical</u>
A Wood Plank	0.092 x 0.289 x 3.66	115	41	272	191
B 6" Sch. 40 Pipe	0.168D x 4.58	287	34	171	119
C 1" Steel Rod	0.0254D x 0.915	8.8	0.79	167	167
D Utility Pole	0.343D x 10.68	1124	143	180	126
E 12" Sch. 40 Pipe	0.32D x 4.58	750	125	154	108
F Automobile	2 x 1.3 x 5	3990	4030	194	136

Missiles A, B, C, and E are to be considered at all elevations and missiles D and F at elevations up to 30 feet above all grade levels within 1/2 mile of the structure.

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

Protection of vital equipment is achieved primarily by separation of redundant safe shutdown systems and by separation of high-energy pipe lines from safe shutdown systems, which are required to be functional following specific pipe rupture events. This redundancy and separation results in a design which requires very few special protective features (such as whip restraints and jet deflectors) to ensure safe shutdown capability following a postulated high-energy line break.

Separation is maintained by barriers such as the containment secondary shield wall, refueling cavity wall and certain primary auxiliary building walls and tunnels or by physical distance. Loadings and jet zones of influence are calculated using methodology described in Section 3.6.2.

3.6.1   POSTULATED PIPING FAILURES IN FLUID SYSTEMS

3.6.1.1   Design Basis

Most systems and components outside containment required for safe plant shutdown are located in the primary auxiliary building. The primary auxiliary building is divided by a structural wall which serves as a barrier between redundant trains of safe shutdown systems and components. Each half of the primary auxiliary building is compartmentalized to separate redundant safe shutdown components to the extent practical. High-energy piping systems located in the auxiliary building, which are not required to be functional for safe shutdown such as steam generator blowdown downstream of containment isolation and auxiliary steam, are routed primarily in designated pipe tunnels in the secondary auxiliary building to provide separation from safe shutdown systems and components. The primary and secondary auxiliary buildings are separated by structural walls that provide physical barriers.

Systems and components inside containment, which are required to be functional for safe plant shutdown, are protected from postulated pipe failure dynamic effects primarily by separation and barriers. The secondary shield wall serves as a barrier between the reactor coolant loops and the containment liner. The refueling cavity walls, the operating floor, and the secondary shield wall provide separation between the reactor coolant loops. The steam generators and pressurizer are enclosed in cavities which also provide separation.

Main steam and main feedwater (downcomer and economizer) lines outside containment are separated from essential systems and components by virtue of the plant arrangement that places these lines in a special enclosure constructed along the roof of the primary auxiliary building. The two sides of this chase adjacent to the primary auxiliary building (floor and one wall) are Seismic Category I concrete walls. The other two sides of this chase are metallic siding adjacent to the environment. In the event of a high-energy line break in this enclosure, the metallic siding will blow out and no significant pressurization will result. The essential portions of these systems (containment isolation valves) are located in the MSIV rooms. These rooms are separated from all other essential systems and components by Seismic Category I concrete slabs and walls.

The NSSS design consists of two steam generators per unit, which facilitates separation of redundant systems and components inside containment. Other than for the safety injection system components, which must circulate cooling water to the vessel, the engineered safety features are generally located outside the secondary shield wall. The safety injection system pipes and cables, which terminate inside the secondary shield wall, are routed outside the secondary shield wall to the extent practical to avoid postulated hazards. Most of the main steam and feedwater piping inside containment is located at higher elevations, and the postulated dynamic effects are separated from safe shutdown systems and components by distance and configuration. Table 3.6-1 provides a list of plant fluid systems that contain high- and moderate-energy piping in the auxiliary and containment buildings.

Table 3.6-2 provides a list of the systems that are required for safe shutdown or to support safe shutdown.

High- and moderate-energy pipe failure locations are postulated as described in Section 3.6.2. Each postulated rupture location is evaluated for its effect on safe shutdown systems and components required following the specific pipe failure event.

3.6.1.1.1 High-Energy Piping Systems

A high-energy pipe failure is postulated in branches or piping runs larger than one inch nominal diameter and which operate during normal plant conditions with high energy fluid.

Included in this category are fluid systems or portions of fluid systems which are pressurized above atmospheric pressure during

normal plant operation and which, in addition, operate during normal plant conditions and where either or both of the following are met:

- A. Maximum operating temperature exceeds 200°F, or
- B. Maximum operating pressure exceeds 275 psig.

Fluid piping systems that qualify as high-energy for only short portions of their operational period are considered moderate-energy systems if the portion of their operational period within the pressure temperature specified above for high energy fluid systems is less than either of the following:

- A. One percent of the normal operating life span of the plant, or
- B. Two percent of the time period required to accomplish its system design function.

In analyzing the effects of a high-energy pipe failure, the consequences of pipe whip, water spray, jet impingement, flooding, compartment pressurization, and environmental conditions are considered.

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3.6.1.1.2 Moderate-Energy Piping Systems

A moderate-energy pipe failure is postulated in branches or piping runs larger than one inch nominal diameter and which operate during normal plant conditions with moderate-energy fluid.

Included in this category are fluid systems or portions of fluid systems which are pressurized above atmospheric pressure during normal plant operation and which, in addition, operate during normal plant conditions and where both of the following are met:

- A. Maximum operating temperature is 200°F or less, and
- B. Maximum operating pressure is 275 psig or less.

In analyzing the effects of a moderate-energy failure, the consequences of water spray, jet impingement, flooding, compartment pressurization, and environmental conditions are considered.

3.6.1.2   Description

A listing of the high-energy lines inside the containment is given in Table 3.6-3. A listing of high-energy lines outside the containment is given in Table 3.6-4. Since the Turbine and Radwaste Buildings contain no safety-related equipment, high-energy line breaks in those buildings are generally excluded from this table.

Essential systems are those systems that are needed to safely shut down the reactor or mitigate the consequences of a pipe break for a given postulated piping failure. However, depending upon the type and location of a postulated pipe break, certain safety equipment may not be classified as essential for that particular event.

The essential systems which are to be protected from the effects of postulated piping failures are identified below. These essential systems were selected for each postulated break to satisfy the protection criteria given in the introduction to Section 3.6.

- A. The following systems, or portions of these systems, are required to mitigate the consequences of postulated breaks of high-energy reactor coolant pressure boundary piping that result in a loss-of-coolant-accident (LOCA) assuming a loss of offsite power.
1. Reactor Protective System.
  2. Engineered Safety Features Actuation System.
  3. Safety Injection System.
  4. Containment Spray System.
  5. Class 1E Electrical Systems, AC and DC (including switchgear, batteries, and distribution systems), 1E cabling and sensing lines.
  6. Diesel Generator Systems, including Diesel Generator Starting, Lubrication, and Combustion Air Intake and Exhaust Systems.
  7. Diesel Fuel Oil Storage and Transfer System.
  8. Hydrogen Recombiner Systems.
  9. Control Building HVAC System.

10. Component Cooling Water System (portions required for operation of other listed systems).
11. Essential Spray Pond System.
12. Fuel Building HVAC System.
13. Diesel Generator Building HVAC System.
14. Main Control Board (See Tables 7.3-2 and 7.3-14 for systems required).
15. Containment Isolation Systems:
  - a. Penetration assemblies
  - b. Isolation valves
  - c. Equipment hatch
  - d. Emergency personnel hatch
  - e. Personnel lock
  - f. Liner plate
  - g. Test connections
  - h. Piping between penetration assemblies and isolation valves.
16. Ex-core Neutron Monitoring System.
17. Safety-related Radiation Monitors (refer to Section 11.5).
18. Shutdown Cooling System.
19. Essential Chilled Water System.
20. Safety Depressurization System.
21. Emergency Feedwater System.
22. Air Coolers
23. Station Service Water System

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B. The following systems, or portions of these systems are required to mitigate the consequences of postulated breaks in high-energy secondary pressure boundary piping (main steam, main feedwater, blowdown, or emergency feedwater) assuming a loss of offsite power.

1. Reactor Protective System.
2. Engineered Safety Features Actuation System.
3. Safety Injection System.
4. Containment Spray System (for breaks inside the containment only).
5. Main Steam and Feedwater System (from unaffected steam generator out to the containment isolation valves, including the atmospheric steam dump, steam supply to the turbine-driven emergency feedwater pump, and the steam generator blowdown line).
6. Shutdown Cooling System.
7. Class 1E Electrical Systems, AC and DC (including Switchgear, Batteries and Distribution Systems), 1E cabling and sensing lines.
8. Diesel Generator System, including Diesel Generator Starting, Lubrication and Combustion Air Intake and Exhaust Systems.
9. Diesel Fuel Oil Storage and Transfer System.
10. Component Cooling Water System (portions required for operation of other listed systems).
11. Essential Spray Pond System.
12. Control Building HVAC System.
13. Fuel Building HVAC System.
14. Main Control Board (See Tables 7.3-2 and 7.3-14 for systems required).
15. Essential Chilled Water System.
16. Containment Isolation Systems:

- a. Penetration assemblies
  - b. Isolation valves
  - c. Equipment hatch
  - d. Emergency personnel hatch
  - e. Personnel lock
  - f. Liner plate
  - g. Test connections
  - h. Piping between penetration assemblies and isolation valves.
- 17. Diesel Generator Building HVAC System.
  - 18. Emergency Feedwater System.
  - 19. Condensate Storage System.
  - 20. Ex-core Neutron Monitoring.
  - 21. Station Service Water System.
  - 22. Air Coolers.
- C. For other postulated breaks not included in items A and B above, systems must not be affected such that any break, evaluated on a case-by-case basis, violates the following criteria:
- 1. The pipe break must not cause a reactor coolant, steam, or feedwater line break.
  - 2. The function of safety systems required to perform protective actions to mitigate the consequences of the postulated break must be maintained.
  - 3. The ability to place the plant in a safe shutdown condition must be maintained.

A systematic approach of multidiscipline analyses of safety-related and associated systems was initiated to verify compliance with design criteria, interface requirements, and safety design bases. On-going reviews of the model identified potential hazards and highlighted susceptibility of essential

equipment from common mode failure, as well as provided an independent method of verification of the availability of essential equipment required to mitigate the consequences of postulated accident scenarios. The resolution of comments raised during these reviews resulted in changes to equipment layout, design of pipe whip and jet impingement restraints, upgrading some non-seismic supports to seismic, and the addition of curbs, drains, and other flood mitigation measures.

The potential effects of flooding as a consequence of a pipe break, or leakage or through-wall cracks (as defined in Sections 3.6.2.1.2.C and 3.6.2.1.2.D) were analyzed on a case-by-case basis to ensure that the operability of safety-related equipment would not be impaired.

An analysis of the potential effects of missiles is discussed in Section 3.5.

The potential environmental effects of steam on essential systems are discussed in Section 3.11. In general, because of the protective measures of redundancy and separation between systems and trains, the consequential effect of the transport of steam will not be sufficient to impair the ability of the essential system to shut down the plant and/or mitigate the consequences of the given accident of interest.

There are no high-energy lines in the vicinity of the control room. As such, there are no effects upon the habitability of the control room by pipe break either from pipe whip, jet impingement, or transport of steam. Further discussion on control room habitability systems is provided in Section 6.4.

3.6.1.3   Safety Evaluation

By means of design features such as separation, barriers, and pipe whip and jet impingement restraints, all of which are discussed below, the effects of pipe break will not damage essential systems to an extent that would impair their design function nor affect necessary component operability.

The ability of specific safety-related systems to withstand a single active failure concurrent with a postulated event is discussed in the failure modes and effects analyses provided in Sections 5.4.7, 6.2, 6.3, 6.5, 7.2, 7.3, 8.3, 9.2 and 10.4.

A.   Separation

The plant arrangement provides separation to the extent practical between redundant safety systems in order to

prevent loss of safety function as a result of hazards different from those for which the system is required to function, as well as for the specific event for which the system is required to be functional. Separation between redundant safety systems with their related auxiliary supporting features is the basic protective measure.

In general, layout of the facility followed a multistep process to ensure adequate separation.

1. Safety-related systems are located away from most high-energy piping.
2. Redundant (e.g., "A" and "B" trains) safety systems and subsystems are located in separate compartments.
3. As necessary, specific components are enclosed to maintain the redundancy required for those systems that must function as a consequence of specific piping failure events.

B. Barriers-Shields and Enclosures

Protection requirements are met through the protection afforded by the walls, floors, columns, abutments, and foundations in many cases. Where adequate protection does not already exist due to separation, additional barriers, deflectors, or shields are provided as necessary to meet the functional protection requirements. Where compartments, barriers, and structures are required to provide the necessary protection, they are designed to withstand the combined effects of the postulated failure plus normal operating loads plus earthquake loadings. E

C. Piping Restraint Protection

Where adequate protection does not already exist due to separation, barriers or shields, piping restraints are provided as necessary to meet the functional protection requirements. Restraints are not provided when it can be shown that the pipe break would not cause unacceptable damage to essential systems or components.

The design criteria for pipe whip restraints are given in Section 3.6.2.3.3.

D. Facility Response Analyses

An analysis of postulated pipe break events was performed to identify those safety-related systems and components that

provide protective actions required to mitigate, to acceptable limits, the consequences of the postulated pipe break event.

Whenever the separation inherent in the plant design is shown to assure the functional capability of the safety systems required following a postulated pipe break event, no additional protective measures are required for that event, an additional considerations of break type, location, orientation, restraints, an other protective measures are not required. When necessary, additional protective measures are incorporated into the design, as required, to assure the functional capability of safety systems required following the postulated pipe break event.

In conducting the facility response analyses, the following criteria are utilized to establish the integrity of systems and components necessary for safe reactor shutdown and maintenance of the shutdown condition:

1. Offsite power is assumed to be unavailable if an automatic turbine generator trip or automatic reactor trip is a direct consequence of a postulated piping failure.
2. In addition to the postulated pipe failure and its accompanying effects, a single active component failure is assumed in the systems required to mitigate the consequences of the postulated piping failure.

The single active component failure is assumed, except as noted in Section 3.6.1.3.D.4.

3. Each high- or moderate-energy fluid system pipe failure is considered separately as a single postulated initiating event occurring during normal plant conditions.
4. Where a postulated piping failure is assumed in one of two redundant trains of a system that is required to operate during normal plant conditions as well as to shut down the reactor, single failures that prevent the functioning of the other train or trains of that system are not assumed, provided the system is designed to Seismic Category I standards, is powered from offsite and onsite sources, and is designed, constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety class systems.

5. All available systems and components, including non-Seismic Category I and those actuated by operator actions, may be employed to mitigate the consequences of a postulated piping failure. In judging the availability of such systems and components, account is taken of the postulated failure and its direct consequences, such as unit trip and loss of offsite power, and of the assumed single active component failure and its direct consequences. The feasibility of carrying out operator actions is based on a minimum of 30 minutes delay responding to alarm indication and adequate access to equipment being available for the proposed actions. (Access to the containment post-LOCA is not assumed.)
6. Piping systems containing high-energy fluids are designed so that the effects of a single postulated pipe break cannot, in turn, cause failures of other pipes or components with unacceptable consequences.
7. For a postulated pipe failure, the escape of steam, water, and heat from structures enclosing the high-energy fluid containing piping does not preclude:
  - a. Accessibility to surrounding areas important to the safe control of reactor operations.
  - b. Habitability of the control room.
  - c. Ability of instrumentation, electric power supplies, and components and controls to initiate, actuate, and complete a safety action. (A loss of redundancy is permissible, but not the loss of function.)

The design criteria define acceptable types of isolation for safety-related elements and for high-energy line from similar elements of the redundant train. Separation is accomplished by:

- A. Routing the two groups through separate compartments, or
- B. Physically separating the two groups by a specified minimum distance, or
- C. Separating the two groups by structural barriers.

The design criteria assure that a postulated failure of a high-energy line or a safety-related element cannot take more than one safety-related train out of service. The failure of a component or subsystem of one train may cause failure of another

portion of the same train; for example, a "B" train high-energy pipe may cause failure of a "B" train electrical tray, but not failure of any "A" train component. The capability to shut the plant down safely under such a failure will therefore remain intact.

Given the separation criteria above, and the pipe break criteria in Section 3.6.2.1.2, the effects of high-energy pipe breaks are not analyzed where it is determined that all essential systems, components, and structures are sufficiently physically remote from a postulated break in that piping run.

3.6.2 DETERMINATION OF BREAK LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

Described herein are the design bases for locating breaks and cracks in piping inside and outside containment, the procedure used to define the thrust at the break location, the jet impingement loading criteria, and the dynamic response models and results.

3.6.2.1 Criteria Used to Define Break and Crack Locations and Configurations

3.6.2.1.1 General Requirements

Postulated pipe ruptures are considered in all plant piping systems and the associated potential for damage to required systems and components is evaluated on the basis of the energy in the system. System piping is classified as high-energy or moderate-energy, and postulated ruptures are classified as circumferential breaks, longitudinal breaks, leakage cracks, or through-wall cracks. Each postulated rupture is considered separately as a single postulated initiating event.

For each postulated circumferential and longitudinal break, an evaluation is made of the effects of pipe whip, jet impingement, compartment pressurization, environmental conditions, and flooding. Also, if required to demonstrate safe plant shutdown, an internal fluid system load evaluation is performed on the effects of fluid forces on components within or bounding the fluid system. For each postulated leakage crack, an evaluation is made of the effects of compartment pressurization, environmental conditions and flooding. For each postulated through-wall crack, an evaluation is made of the effects of environmental conditions and flooding. The evaluation of the required systems and components demonstrate that the protection requirements of Section 3.6.1 are met.

Irrespective of the fact that the criteria in Section 3.6.2 may not require specific breaks, if a structure outside containment separates a high-energy line from an essential component, that separating structure is designed to withstand the consequences of the pipe break in the high-energy line that produces the greatest effect at the structure.

3.6.2.1.2 Postulated Rupture Descriptions

A. Circumferential Break

A circumferential break is assumed to result in pipe severance with full separation of the two severed pipe ends unless the extent of separation is limited by consideration of physical means. The break plane area (A_e) is assumed perpendicular to the longitudinal axis of the pipe, and is assumed to be the cross-sectional flow area of the pipe at the break location. The break flow area (A_f) from each of the broken pipe segments for a circumferential break, with full separation of the two broken pipe segments, is equal to the break plane area (A_e). The break flow area, discharge coefficient and discharge correlation are substantiated analytically or experimentally.

B. Longitudinal Break

A longitudinal break is assumed to result in a split of the pipe wall along the pipe longitudinal axis, but without severance. The break plane area (A_e) is assumed parallel to the longitudinal axis of the pipe and equal to the cross-sectional flow area of the pipe at the break location. The break flow area (A_f) is equal to the break plane area (A_e). The break is assumed to be circular in shape or elliptical ($2D \times D/2$) with its long axis parallel to the axis. The discharge coefficient and any other values used for the area or shape associated with a longitudinal break are substantiated analytically or experimentally.

C. Leakage Crack

A leakage crack is assumed to be a crack through the pipe wall where the size of the crack and corresponding flow rate are determined by analysis and a leak detection system, as described in Section 3.6.3.

D. Through-Wall Crack

A through-wall crack is assumed to be a circular orifice through the pipe wall of cross-sectional flow area equal to the product of one-half the pipe inside diameter and one-half the pipe wall thickness.

3.6.2.1.3 Piping Evaluated for Leak-Before-Break

A leak-before-break evaluation is performed for Class 1 piping with a diameter of ten inches or greater (i.e., the reactor coolant system (RCS) main loop piping, surge line, shutdown cooling and safety injection lines) and for the main steam line inside containment in order to eliminate the dynamic effects of pipe rupture from the design basis. The evaluation is intended to meet the requirements of 10 CFR 50, Appendix A, General Design Criterion (GDC) 4. The evaluation is performed using the guidelines of NUREG 1061, Vol. 3 (Reference 1) as described in Section 3.6.3.

3.6.2.1.4 Piping Other than Piping Evaluated for Leak-Before-Break

This section applies to all high- and moderate-energy piping other than that whose dynamic effects due to pipe breaks are eliminated from the design basis by leak-before-break evaluation, as identified in Section 3.6.2.1.3.

3.6.2.1.4.1 Postulated Rupture Locations

A. Class 1 Piping

Ruptures, as specified in Items D and E below, are postulated to occur at the following locations in each piping network designed in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III (Reference 2) for Class 1 piping:

1. the terminal ends of the pressurized portions of the network.
2. intermediate locations where S from equation (10) exceeds $2.4S_m$.
3. intermediate locations where U exceeds 0.1.

where, as defined in Subarticle NB-3650,

S = primary-plus-secondary stress-intensity range under the combination of loadings for which either Level A or Level B service limits have been specified, as calculated from equation (10).

S_m = allowable stress-intensity value.

U = the cumulative usage factor.

B. Class 2, Class 3, or Seismically Analyzed ANSI B31.1 Piping

Ruptures, as specified in Items D and E below, are postulated to occur at the following locations in each piping network designed in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, (Reference 2) for Class 2 and Class 3 piping, or with the rules of the ASME Code for Pressure Piping, B31, Power Piping, ANSI/ASME B31.1-1983 (Reference 3) for seismically analyzed ANSI B31.1 piping:

1. the terminal ends of the pressurized portion of the network, and
2. either
  - a. intermediate locations of potential high stress or fatigue such as pipe fittings, valves, flanges and welded-on attachments, or
  - b. intermediate locations where the stress, S , exceeds $0.8(X + Y)$.

where, as defined in Subarticle NC-3650,

S = stresses under the combination of loadings for which either Level A or Level B service limits have been specified, as calculated from the sum of equations (9) and (10).

X = equation (9) Service Level B allowable stress.

Y = equation (10) allowable stress.

E

C. Non-Seismically Analyzed ANSI B31.1 Piping

Ruptures, as specified in Items D and E, are postulated to occur at the following locations in each ASME Code for Pressure Piping, B31, Power Piping, ANSI/ASME B31.1-1983 (Reference 3) piping network that is not seismically analyzed.

1. at terminal ends of the pressurized portions of the network, and
2. at each intermediate location of potential high stress or fatigue, such as pipe fittings, valves, flanges, and welded-on attachments.

D. Break Locations

Both circumferential and longitudinal breaks are postulated to occur, but not concurrently, in all high-energy piping systems at the locations specified in Items A, B, or C, except as follows:

1. Circumferential breaks are not postulated in piping runs of a nominal diameter equal to or less than 1 inch.
2. Longitudinal breaks are not postulated in piping runs of a nominal diameter less than 4 inches.
3. Longitudinal breaks are not postulated at terminal ends.

4. Only one type of break is postulated at locations where, from a detailed stress analysis, such as finite-element analysis, the state of stress can be used to identify the most probable type. If the primary plus secondary stress in the axial direction is found to be at least 1.5 times that in the circumferential direction for the most severe loading combination associated with Level A and Level B service limits, then only a circumferential break is postulated. Conversely, if the primary plus secondary stress in the circumferential direction is found to be at least 1.5 times that in the axial direction for the most severe loading combination associated with Level A and Level B service limits, then only a longitudinal break is postulated.
5. Circumferential and longitudinal breaks are not postulated at locations where the requirements of Item F are satisfied.
6. Circumferential and longitudinal breaks are not postulated at locations where the criterion in Item E.2 is used.

E. Crack Locations

1. Through-Wall Cracks

Through-wall cracks are postulated in all high-energy and moderate-energy piping systems having a nominal diameter greater than 1 inch at the locations specified in A, B or C, except that through-wall cracks are not postulated at locations where:

- a. For Class 1 piping, the calculated value of S, as defined in Item A, is less than one-half the limits of Item A.2.b.
- b. For Class 2, Class 3 or seismically analyzed ANSI B31.1 piping, the calculated values of S as defined in Item b is less than one-half the limits of Item B.2.b.
- c. The requirements of Item F are satisfied.
- d. The criterion in 2. below is used.

2. Leakage Cracks

A leakage crack is postulated in place of a circumferential break, or longitudinal break, or through-wall crack, if justified by an analysis performed on the pipeline in accordance with the requirements of Section 3.6.3.

F. Piping Near Containment Isolation Valves

Ruptures are not postulated between the containment wall and the inboard or outboard isolation valves in piping, which is designed in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III (Reference 2), and which meets the following additional requirements:

1. The limits for postulating intermediate rupture locations, as specified in Item A.2.b for Class 1 piping and Item B.2.b for Class 2 and 3 piping, are not exceeded in that portion of piping.
2. Following a postulated pipe break of high-energy piping beyond either isolation valve, the stresses in the piping from the containment wall, to and including the length of the isolation valve, are maintained within Level C Service Limits as specified in the ASME Boiler and Pressure Vessel Code, Section III, (Reference 2).
3. The design and in-service inspection requirements, as specified in MEB 3-1 (Reference 4), are satisfied.
4. The containment isolation valves are appropriately qualified to assure that operability and leak tightness are maintained when subjected to any combination of loadings, which may be transmitted to the valves from postulated pipe breaks beyond the valves.

3.6.2.1.4.2 Postulated Rupture Configurations

A. Break Configurations

Where the postulated break location is at a tee, elbow, or the following pipe locations, the configurations and types of breaks are determined as follows:

1. Without the benefit of a detailed stress analysis, the following are assumed:

- a. Circumferential breaks are postulated to occur individually at each tee or elbow pipe-to-fitting weld where the criteria in Section 3.6.2.1.4.1, Item C are exceeded, and longitudinal breaks postulated to occur individually on each side of the tee or elbow at its center and oriented perpendicular to the plane of the fitting.
  - b. At a branch run connection, a circumferential break is postulated at the branch run-to-main run weld, or the branch run-to-fitting weld, and the break plane area (A_e) is assumed to be the cross-sectional flow area of the branch.
  - c. At a welded attachment (lug, stanchion, etc.) a longitudinal break is postulated at the centerline of the welded attachment with an area equal to the pipe surface area that is bounded by the attachment weld.
  - d. At an axisymmetric pipe location, such as a reducer, circumferential and longitudinal breaks are postulated at each pipe-to-fitting weld where the criteria in Section 3.6.2.1.4.1, Item C are exceeded. Longitudinal breaks are oriented to produce out-of-plane bending of the piping configuration.
2. Alternatively, where a detailed stress analysis or test is performed, the results are used to predict the most probable rupture location(s) and type of break.

B. Crack Configurations

At a postulated leakage crack or through-wall crack location, the orifice is assumed to be located noncurrently at each and every point about the circumference of the pipe, unless otherwise substantiated.

3.6.2.1.5 Details of Containment Penetrations

Details of containment penetrations are discussed in Sections 3.8.1 and 3.8.2.

3.6.2.2   Analytical Methods to Define Forcing Functions and Response Models**3.6.2.2.1   Piping Evaluated for Leak-Before-Break**

There are no forcing functions or response models for the reactor coolant loop, surge line, shutdown cooling line, safety injection line and main steam line based upon elimination of dynamic effects by leak-before-break evaluation.

3.6.2.2.2   Analytical Methods to Define Forcing Functions and Response Models for Piping Excluding that Evaluated for Leak-Before-Break

This section applies to all high-energy piping other than that whose dynamic effects due to pipe breaks are eliminated from the design basis by leak-before-break evaluation.

3.6.2.2.2.1   Determination of Pipe Thrust and Jet Loads**A.   Circumferential Breaks**

Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections, unless physically limited by piping restraints, structural members, or piping stiffness. The dynamic force of the jet discharge at the break location is based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically determined thrust coefficient. Limited pipe displacement at the break locations, line restriction flow limiters, positive pump controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of the jet discharge. Pipe whip is assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction.

B.   Dynamic Force of the Fluid Jet Discharge

The dynamic force of the fluid jet discharge is based on a circular break area equal to the cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically determined thrust coefficient, as determined for a circumferential break at the same location. Line restrictions, flow limiters,

positive pump-controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of jet discharge.

Piping movement is assumed to occur in the direction of the jet reaction, unless limited by structural members, piping restraints, or piping stiffness.

C. Pipe Blowdown Force and Wave Force

The fluid thrust forces that result from either postulated circumferential or longitudinal breaks, are calculated using a simplified one step forcing function methodology. This methodology is based on the simplified methods described in ANSI 58.2 (Reference 5) and in Reference 6.

When the simplified method discussed above leads to impractical protective measures, then a more detailed computer solution which more accurately reflects the postulated pipe rupture event is used. The computer solution is based on the NRC's computer program developed for calculating two-phase blowdown forces (Reference 7).

D. Evaluation of Jet Impingement Effects

Jet impingement force calculations are performed only if structures or components are located near postulated high energy line breaks and it cannot be demonstrated that failure of the structure or component will not adversely affect safe shutdown capability.

3.6.2.2.2.2 Methods for the Dynamic Analysis of Pipe Whip

Pipe whip restraints usually provide clearance for thermal expansion during normal operation. If a break occurs, the restraints or anchors nearest the break are designed to prevent unlimited movement at the point of break (pipe whip). A finite difference model will be used to analyze simplified models of the local region near the break. Displacements and strains of the pipe and restraint will be estimated using a power law moment curvature relationship.

A. Finite Difference Analysis

A finite difference formulation specialized to the case of a straight beam and neglecting axial inertia and large deflection effects is used for the analysis of pipe whip. The dynamic analysis is performed by direct numerical time integration of the equations of motion presented in Appendix 3.6A.

3.6.2.2.2.3 Method of Dynamic Analysis of Unrestricted Pipes

The impact velocity and kinetic energy of unrestricted pipes is calculated on the basis of the assumption that the segments at each side of the break act as rigid-plastic cantilever beams subject to piecewise constant blowdown forces. The hinge location is fixed either at the nearest restraint or at a point determined by the requirement that the shear at an interior plastic hinge is zero. The kinetic energy of an accelerating cantilever segment is equal to the difference between the work done by the blowdown force and that done on the plastic hinge. The impact velocity V is found from the expression for the kinetic energy:

$$KE = (1/2) M_{eg} V_I^2$$

where M_{eg} is the mass of the single degree of freedom dynamic model of the cantilever. The impacting mass is assumed equal to M_{eg} .

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.3.1 Pipe Whip Restraints and Jet Deflectors for Piping Evaluated for Leak-Before-Break

There are no pipe whip restraints and jet deflector for the reactor coolant loop, surge line, shutdown cooling line, safety injection line and main steam line based upon elimination of dynamic effects due to pipe breaks by leak-before-break evaluation.

3.6.2.3.2 Pipe Whip Restraints and Jet Deflectors for Piping Other than that Evaluated for Leak-Before-Break

This section applies to pipe whip restraints for all piping other than that whose dynamic effects due to pipe breaks are eliminated from the design basis by leak-before-break evaluation.

3.6.2.3.2.1 General Description of Pipe Whip Restraints

Pipe whip restraints are provided to protect the plant against the effects of whipping during postulated pipe break. The design of pipe whip restraints is governed not only by the pipe break blowdown thrust, but also by functional requirements, deformation limitations, properties of whipping pipe and the capacity of the support structure. Typically, a pipe whip restraint consists of a ring around the pipe and components supporting the ring from

the supporting structure. The diameter of the ring is established considering the pipe diameter, maximum thermal movement of pipe, thickness of insulation and an additional 1/2-inch for installation tolerance. The restraint is designed for the impact force induced by the maximum possible initial gap between the ring and the pipe.

This impact energy is usually too high for an elastic restraint system or support structure to absorb. Therefore, energy absorbing measures designed by the energy balance approach (impact energy + external work = internal energy of pipe restraint system), are provided.

3.6.2.3.2.2 Pipe Whip Restraint Components

Pipe whip restraints typically consist of the following components:

A. Energy Absorbing Members

Members that are under the influence of impacting pipes (pipe whip) absorb energy by significant plastic deformations (e.g., rods, and crushable honeycomb material).

B. Non-Energy Absorbing Members

Those components which form a direct link between the pipe and the structure (e.g., ring and components other than energy absorbing members).

C. Structural Attachments

Those fasteners which provide the method of attaching connecting members to the structure or the ring (e.g., weld attachments).

D. Structural Components

Steel and concrete support structures which ultimately carry the restraint load. Design criteria are specified in Section 3.8.

3.6.2.3.2.3 Design Loads

Restraint design loads, the reactions, and the corresponding deflections are established using the criteria delineated in Section 3.6.2.2.2.

3.6.2.3.2.4 Allowable Stresses

The allowable stresses are as follows:

- A. For energy absorbing members: $0.95 F_{yd}$ with $0.5 \epsilon_u$ strain for steel in tension, where the dynamic yield strength, F_{yd} is considered 15% higher than the static yield strength F_{ys} and ϵ_u is the ASTM specified minimum elongation for the rod given as a percentage with 50% usable strain for crushable energy absorption material in compression as determined by dynamic testing.
- B. For non-energy absorbing member: 1.6 times the AISC allowable stress, but not to exceed $0.95 F_{yd}$ for bending and $0.55 F_{yd}$ for shear where F_{yd} is considered 10% higher than F_{ys} for compression members, the allowable stress is 0.9 times the buckling stress F_{CRC} as follows:

$$F_{CRC} = 5/3 \times F_a \times DIF$$

where

$5/3$ = Lower bound factor of safety in AISC for compression stress

F_a = AISC allowable compression stress

DIF = Dynamic increase factor = 1.1

- C. For structural attachments and structural components - allowable stresses will be the same as described in Sections 3.8.3 and 3.8.4.

3.6.2.3.2.5 Design Criteria

The unique features in the design of pipe whip restraint components relative to the structural steel design are geared to the loads used and the allowable stresses. These are as follows:

- A. Energy-absorbing members are designed for the restraint reaction and the corresponding deflection established according to the pipe size and material and the blowdown force using the criteria delineated in Section 3.6.2.2.
- B. Non-energy-absorbing members and their attachments are designed for 1.25 times the restraint reaction to ensure that the required deflection occurs in the energy absorbing members and that the connecting members remain elastic.

- C. Structural components and their attachment to the building structure are designed for the restraint reaction with an appropriate dynamic load factor.

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All essential components are evaluated for jet impingement and pipe whip effects using a dynamic or an equivalent static analysis of testing to demonstrate either the functional capability and/or operability in addition to the structural integrity of the component.

3.6.2.3.2.6 Materials

The materials used are as follows:

- A. For energy-absorbing members: ASTM A-1093 Grade B7 or equivalent for tension rods, and crushable honeycomb made of stainless steel for compression.
- B. For other components: ASTM A-588, ASTM A-572 Grade 50, and ASTM A-36. Charpy tests will be performed on steels subjected to impact loads and lamination tests are performed on steels subjected to through thickness tension.

3.6.2.3.2.7 Jet Impingement Shields

Protection from jets is provided by using separation and redundancy, as described in Section 3.6.1, in lieu of jet shields.

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3.6.2.4 Guard Pipe Assembly Design Criteria

Guard pipes to limit pressurization effects in the containment penetration area will not be used.

3.6.3 LEAK-BEFORE-BREAK EVALUATION PROCEDURE

This section describes Leak-Before-Break (LBB) analysis to all applicable piping. LBB analysis is used to eliminate, from the structural design bases the dynamic effects of double-ended guillotine breaks and equivalent longitudinal breaks for an applicable piping system.

3.6.3.1 Applicability of LBB

Piping evaluated for LBB is first shown to meet the applicability requirements for NUREG 1061, Volume 3. The piping is designed to meet the requirement to be not particularly susceptible to

failure from the effects of corrosion, water hammer or low- and high-cycle fatigue, or degradation or failure of the piping from indirect causes.

3.6.3.2   Leakage Crack Location

A survey of the piping is performed to determine the locations of highest stress loading and coincident poorest material properties. All base metal, weld materials, heat affected zones in the vicinity of the terminal ends, and all intermediate elbow locations are considered.

3.6.3.3   Leak Detection

There are two major aspects to leak rate based on crack detection in addition to the crack opening size; leak detection capability, and flow rate correlation for leakage through a crack.

3.6.3.3.1   Leak Detection System

A leak detection system is recommended by Regulatory Guide 1.45, Reference 8, capable of detecting a leakage rate of less than 1.0 gpm from the primary system. NUREG-1061, Volume 3, recommends a safety margin of ten on the leak detection system. Diverse measurement means are provided, including water inventory monitoring, sump level and flow monitoring, and measurement of airborne radioactive particulates or gases (see Section 5.2.5). Leak detection system requirements to support the LBB analysis for main steam line piping are met by a combination of humidity detectors, air coolers, radioactive airborne activity sensors and sump flow and level meters.

3.6.3.3.2   Flow Rate Correlation

The other major aspect of crack detection based on the leak rate, namely the flow rate correlation for leakage through a given crack size, cannot be predicted precisely. Variables such as surface roughness of the side walls of the crack, the nonparallel relationship of the side walls due to the elongated crack shape, and possibly zigzag tearing of the material during crack formation all introduce uncertainties in defining an exact flow rate correlation.

The leakage rate required to be detectable is 1.0 gpm. The licensing guidelines (NUREG 1061, Volume 3) recommend a factor of 10 on that leakage rate, for conservatism; therefore, a crack length which leaks 10 gpm at normal operating conditions is selected as the design leakage crack. Using recent work by EPRI (Reference 10), the leakage rate per square inch of leak area in the 10 gpm leakage rate range is computed to be approximately 250

gpm per square inch for the primary system for the range of pipe sizes of interest. The crack opening area corresponding to the 10 gpm rate is found to be 0.04 square inch. This crack opening area is used to determine the length of the detectable crack for stability evaluation.

3.6.3.4 Screening of Leakage Crack Sizes Using EPRI/GE Estimation Scheme

Prior to detailed calculations of through-wall leakage cracks and corresponding margins on loads and crack sizes, a preliminary scoping evaluation is performed. In this part, all possible locations in the piping evaluated are screened to identify the most critical candidates for detailed study. The screening study is performed using the EPRI/GE estimation scheme (Reference 11) for the determination of crack opening areas using elastic plastic fracture mechanics methods, and the C-E developed JEST computer program for the leakage rates through cracks.

This estimation procedure is used to compare the severity of hypothesized flaws in all piping locations in order to reduce the number of cases to be subjected to detailed analysis. The procedure also provides an estimate of the leakage crack length for input to the detailed finite-element analysis discussed in Section 3.6.3.1.6.

3.6.3.5 Material Properties

For the main coolant loop, the hot and cold leg piping material is SA516 Gr70. All hot- and cold-leg pipe-to-pipe welds and the pipe-to-reactor vessel, steam generator and reactor coolant pump safe end welds are carbon steel. All main loop component nozzles are SA508 CL 2 or 3 carbon steel or SA541 CL 1, 2 or 3. The surge line is SA351 GR CF8M stainless steel, resulting in bimetallic safe end welds. The shutdown cooling line and the direct vessel safety injection line are both Type 304 stainless steel. The main steam line is SA516 Gr70.

The detailed analysis of cracks in pipe welds requires consideration of the properties of the pipe and the weld materials. Previous work by C-E has shown that a conservative bounding analysis results when the material stress-strain properties of the base metal (lower yield) and the fracture properties of the weld (lower toughness) are used for the entire structure, (Reference 12). This material representation is used for all analyses. The tensile (stress-strain) curves and the J_D vs. Δa curves are required for each material type.

3.6.3.6   Leakage Crack Size Determination

It is necessary that hypothesized through-wall cracks open significantly to allow detection by normal leakage monitoring under normal full power loadings. The crack opening area is calculated at all the chosen locations for normal operating conditions, since these are the most prevalent conditions during plant operating life, when leakage is to be detected. A range of crack sizes is considered to evaluate the sensitivity to leakage rate assumptions. The result of the analysis of several assumed crack sizes is then used to estimate the leakage length. Under normal operation loading, almost all the material in the pipe is in the elastic range. With a small amount of plasticity, this method is very accurate in estimating the crack size. The crack opening area is computed on the inside and outside of the pipe and the smaller area is used.

The finite-element analysis is performed for those cracks determined by the estimation scheme to be of the assured leak detection size. The analysis is also performed for a crack twice that length subjected to the same loading conditions. These two analysis results enable a curve of crack opening area vs. crack length to be drawn. From this curve, the length of the assured leak detectable crack is determined.

3.6.3.7   Computation of J-Integral Values

3.6.3.7.1   Range of Crack Sizes

The crack lengths estimated using the EPRI/GE estimating scheme (Reference 11) are input to the detailed stability analysis of the through-wall cracks in the piping evaluated. The finite-element analysis is carried out for the estimated leakage crack size and twice that length. This procedure, therefore, considers the stability of a range of crack lengths for all locations selected for the analysis.

3.6.3.7.2   J-Integral

The stability of through-wall cracks is evaluated using the J-integral technique. The J-integral is determined in the finite-element analysis for pressure, normal operation, and safe shutdown earthquake loadings for two different crack lengths for each geometric model. For the margin on loads evaluation, the J-integral for the estimated leakage crack size is also evaluated for $\sqrt{2} \times (\text{Pressure} + \text{NOP} + \text{SSE})$ loads. A typical J-integral vs. load increment curve is shown in Figure 3.6-1.

3.6.3.8   Stability Evaluation

The stability of the cracked pipes is assessed by comparing the J-integral value due to the applied loads on the pipe to the material crack resistance. The stability criterion for ductile crack extension employed is:

if $J_{\text{applied}} < J_{\text{IC material}}$, or

$\frac{dJ}{da} \text{ applied} < \frac{dJ}{da} \text{ material}$

then crack stability is assured.

The slope of the J-integral vs. a (crack length) curve for each location is obtained by fitting a polynomial curve through the computed J-integral values. By plotting J-integral vs. dJ/da , then both parts of the stability criterion can be evaluated simultaneously.

The evaluations are performed for the locations chosen to envelop all limiting cases. The pipes with the leakage crack length subject to loads of $\sqrt{2} \times (P+NOP+SSE)$ and the pipes with crack length twice the leakage crack length with loads of $(P+NOP+SSE)$ are demonstrated to have significant margin between the material curve and the loading curve, indicating that all pipe locations satisfy the LBB crack stability criteria.

3.6.3.9   Results

Class 1 piping with a diameter of ten inches or greater and main steam line piping meet all the criteria for application of the leak-before-break according to NUREG 1061, Volume 3. Specifically, these criteria require that:

- A. Cracks which are assumed to grow through the pipe wall leak significantly while remaining stable. The amount of leakage is detectable with a safety margin of at least a factor of 10.
- B. Cracks of the length that leak at the rate in A. can withstand normal operation plus safe shutdown earthquake loads with a safety factor of at least $\sqrt{2}$.
- C. Cracks twice as long as those addressed in B. will remain stable when subjected to normal operation plus safe shutdown earthquake loads.

REFERENCES FOR SECTION 3.6

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3. ASME Code for Pressure Piping, B31, Power Piping, ANSI/ASME B31.1-1986 Edition.
4. USNRC Branch Technical Position MEB 3-1 - Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment, attached to Standard Review Plan 3.6.2, July 1981.
5. American National Standard Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture, ANSI/ANS 58.2-1988.
6. R. T. Lahey, Jr. and F. J. Moody, "Pipe Thrust and Jet Loads," The Thermal Hydraulics of a Boiling Water Nuclear Reactor, Section 9.2.3, pp. 375-409, Published by American Nuclear Society, Prepared by the Division of Technical Information United States Energy Research and Development Administration, 1977.
7. RELAP 4/MOD 5, Computer Program User's Manual 098. 026-5.5.
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9. NUREG/CR-1319, "Cold Leg Integrity Evaluation," Battelle Columbus Laboratories, February 1980.
10. PICEP: Pipe Crack Evaluation Program, EPRI NP 3596-SR, August, 1984.
11. "An Engineering Approach for Elastic-Plastic Fracture Analysis," EPRI NP2931, by V. Kumar, M. D. German, C. F. Shih, July 1981.
12. "Analysis of Cracked Pipe Weldments," EPRI NP-5057, February 1987.

TABLE 3.6-1

HIGH- AND MODERATE-ENERGY FLUID SYSTEMS

High-Energy Fluid Systems*

Main Steam
Main Feedwater
Steam Generator Blowdown
Auxiliary Steam
Reactor Coolant
Surge Line
Safety Injection
Chemical and Volume Control

Moderate-Energy Fluid Systems

Chemical and Volume Control
Component Cooling
Steam Generator Blowdown
Safety Injection/Shutdown Cooling
Containment Air Monitoring/Sampling
Diesel Fuel Oil
Station Service Water
Fire Protection
H₂, N₂, and CO₂
Service Water²
Spent Fuel Pool Cooling and Cleanup
Station Heating
Containment Spray
Drain
Chilled Water
Breathing Air
Instrument Air
Service Air

E

* Systems shown are either totally or partially high-energy.

TABLE 3.6-2

SYSTEMS REQUIRED FOR SAFE SHUTDOWN AND/OR TO MITIGATE THE CONSEQUENCES
OF A DESIGN-BASIS ACCIDENT*

Reactor Protective
Reactor Coolant
Safety Injection/Shutdown Cooling
Containment Spray
Emergency Feedwater
Component Cooling Water
Station Service Water
Auxiliary Power
Area Radiation
Containment Isolation
Condensate Storage
Battery and DC Distribution
Diesel Generator
Diesel Fuel Oil
Engineered Safety Features
Combustible Gas Control
Instrument and Control Power
In-core Thermocouple
Ex-core Neutron Monitoring
Main Steam
Main Feedwater
Control Element Assembly Drive
Control Room HVAC
Diesel Generator Room HVAC
ESF Switchgear Room HVAC
Station Service Water Pump House HVAC
Primary Containment Ventilation
Safety Injection System Equipment Room HVAC
Essential Chilled Water
Essential Spray Pond
Main Control Board
Safety Depressurization

* Systems listed are either totally or partially required for safe shutdown.

TABLE 3.6-3

(Sheet 1 of 11)

HIGH-ENERGY LINES WITHIN CONTAINMENT

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.
1	Main Steam	From SG No. 1 Line #1 to Cont Penetration	Yes	Yes	10.1-2
2	Main Steam	From SG No. 1 Line #2 to Cont Penetration	Yes	Yes	10.1-2
3	Main Steam	From SG No. 2 Line #1 to Cont Penetration	Yes	Yes	10.1-2
4	Main Steam	From SG No. 2 Line #2 to Cont Penetration	Yes	Yes	10.1-2
5	SG Blowdown	SG No. 1 Blowdown Line #1 to Common Blowdown Line	Yes	Yes	10.1-2, 10.4.8-1
6	SG Blowdown	SG No. 1 Blowdown Line #2 to Common Blowdown Line	Yes	Yes	10.1-2, 10.4.8-1
7	SG Blowdown	SG No. 1 Blowdown Common Line to Cont Penetration	Yes	Yes	10.1-2, 10.4.8-1
8	SG Blowdown	SG No. 2 Blowdown Line #1 to Common Blowdown Line	Yes	Yes	10.1-2, 10.4.8-1
9	SG Blowdown	SG No. 2 Blowdown Line #2 to Common Blowdown Line	Yes	Yes	10.1-2, 10.4.8-1

TABLE 3.6-3 (Cont'd)

(Sheet 2 of 11)

HIGH-ENERGY LINES WITHIN CONTAINMENT

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.
10	SG Blowdown	SG No. 2 Blowdown Common Line to Cont Penetration	Yes	Yes	10.1-2, 10.4.8-1
11	SG Wet Layup Recirc	SG No. 1 Wet Layup Recirc Nozzle to Interior Check Valve	Yes	Yes	10.1-2, 10.4.8-1
12	SG Wet Layup Recirc	SG No. 2 Wet Layup Recirc Nozzle to Interior Check Valve	Yes	Yes	10.1-2, 10.4.8-1
13	Main Feedwater	To SG No. 1 Economizer Nozzle #1 from Economizer Common Feedwater Line	Yes	Yes	10.1-2
14	Main Feedwater	To SG No. 1 Economizer Nozzle #2 from Economizer Common Feedwater Line	Yes	Yes	10.1-2
15	Main Feedwater	From SG No. 1 Economizer Common Main Feedwater Line to Cont. Penetration	Yes	Yes	10.1-2
16	Main Feedwater	From SG No. 1 Downcomer Nozzle to Cont. Penetration	Yes	Yes	10.1-2
17	Main Feedwater	To SG No. 2 Economizer Nozzle #1 from Economizer Common Feedwater Line	Yes	Yes	10.1-2

TABLE 3.6-3 (Cont'd)**(Sheet 3 of 11)****HIGH-ENERGY LINES WITHIN CONTAINMENT**

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.
18	Main Feedwater	To SG No. 2 Economizer Nozzle #2 from Economizer Common Feedwater Line	Yes	Yes	10.1-2
19	Main Feedwater	From SG No. 2 Economizer Common Main Feedwater Line to Cont. Penetration	Yes	Yes	10.1-2
20	Main Feedwater	From SG No. 2 Downcomer Nozzle to Cont. Penetration	Yes	Yes	10.1-2
21	Reactor Coolant	SG No. 1 RCS Hot Leg Loop	Yes	Yes	5.1.2-1
22	Reactor Coolant	SG No. 1 RCS Pump 1A Discharge	Yes	Yes	5.1.2-1
23	Reactor Coolant	SG No. 1 RCS Pump 1B Discharge	Yes	Yes	5.1.2-1
24	Reactor Coolant	SG No. 1 RCS Pump 1A Suction	Yes	Yes	5.1.2-1
25	Reactor Coolant	SG No. 1 RCS Pump 1B Suction	Yes	Yes	5.1.2-1
26	Reactor Coolant	SG No. 1 RCS Loop 1A Drain Line to RDT (High-Energy to Isolation Valve)	Yes	Yes	5.1.2-1
27	Reactor Coolant	SG No. 1 RCS Loop 1B Drain Line to RDT (High-Energy to Isolation Valve)	Yes	Yes	5.1.2-1

TABLE 3.6-3 (Cont'd)

(Sheet 4 of 11)

HIGH-ENERGY LINES WITHIN CONTAINMENT

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.
28	Reactor Coolant	SG No. 2 RCS Hot Leg Loop	Yes	Yes	5.1.2-1
29	Reactor Coolant	SG No. 2 RCS Pump 2A Discharge	Yes	Yes	5.1.2-1
30	Reactor Coolant	SG No. 2 RCS Pump 2B Discharge	Yes	Yes	5.1.2-1
31	Reactor Coolant	SG No. 2 RCS Pump 2A Suction	Yes	Yes	5.1.2-1
32	Reactor Coolant	SG No. 2 RCS Pump 2B Suction	Yes	Yes	5.1.2-1
33	Reactor Coolant	SG No. 2 RCS Loop 2A Drain Line to RDT (High-Energy to Isolation Valve)	Yes	Yes	5.1.2-1
34	Reactor Coolant	SG No. 2 RCS Loop 2B Drain Line to RDT (High-Energy to Isolation Valve)	Yes	Yes	5.1.2-1
35	Reactor Coolant	SG No. 2 RCS Hot Leg Drain Line to RDT (High-Energy to Isolation Valve RC-215	Yes	Yes	5.1.2-1
36	Reactor Coolant	Pressurizer Relief Line #1 to Pressurizer Safety Valve #1	Yes	Yes	5.1.2-1
37	Reactor Coolant	Pressurizer Relief Line #2 to Pressurizer Safety Valve #2	Yes	Yes	5.1.2-1

TABLE 3.6-3 (Cont'd)**(Sheet 5 of 11)****HIGH-ENERGY LINES WITHIN CONTAINMENT**

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.
38	Reactor Coolant	Pressurizer Relief Line #3 to Pressurizer Safety Valve #3	Yes	Yes	5.1.2-1
39	Reactor Coolant	Pressurizer Relief Line #4 to Pressurizer Safety Valve #4	Yes	Yes	5.1.2-1
40	Reactor Coolant	Pressurizer Spray Line from Loop 1A to 1A Spray Control Valve	Yes	Yes	5.1.2-1
41	Reactor Coolant	Pressurizer Spray Line from Loop 1B to 1B Spray Control Valve	Yes	Yes	5.1.2-1
42	Reactor Coolant	Pressurizer Spray Line from Loop 1A Spray Control Valve to Pressurizer Spray Common Header	Yes	Yes	5.1.2-1
43	Reactor Coolant	Pressurizer Spray Lines from Loop 1B Spray Control Valve to Pressurizer Spray Common Header	Yes	Yes	5.1.2-1
44	Reactor Coolant	Pressurizer Spray Common Header to Pressurizer	Yes	Yes	5.1.2-1
45	Reactor Coolant	Pressurizer Surge Line	Yes	Yes	5.1.2-1
46	Reactor Coolant	Division 1 RCS Hot Leg to SIS Hot Leg Injection Interior Check Valve SI-165	Yes	Yes	5.1.2-1, 6.3.2-1B

TABLE 3.6-3 (Cont'd)

(Sheet 6 of 11)

HIGH-ENERGY LINES WITHIN CONTAINMENT

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.
47	Reactor Coolant	Division 2 RCS Hot Leg to SIS Hot Leg Injection Interior Check Valve SI-166	Yes	Yes	5.1.2-1, 6.3.2-1B
48	Reactor Coolant	Direct Vessel Injection Connection #1 to SIS Interior Check Valve	Yes	Yes	5.1.2-1, 6.3.2-1B
49	Reactor Coolant	Direct Vessel Injection Connection #2 to SIS Interior Check Valve	Yes	Yes	5.1.2-1, 6.3.2-1B
50	Reactor Coolant	Direct Vessel Injection Connection #3 to SIS Interior Check Valve	Yes	Yes	5.1.2-1, 6.3.2-1B
51	Reactor Coolant	Direct Vessel Injection Connection #4 to SIS Interior Check Valve	Yes	Yes	5.1.2-1, 6.3.2-1B
52	Safety Depress System	Branch-Off of Pressurizer Safety Valve #1 Steam Line #1 to SD-1 (Rapid Depress. Line)	Yes	Yes	5.1.2-1, 6.7-1
53	Safety Depress System	Branch-Off of Pressurizer Safety Valve #4 Steam Line #4 to SD-3 (Rapid Depress. Line)	Yes	Yes	5.1.2-1, 6.7-1

TABLE 3.6-3 (Cont'd)

(Sheet 7 of 11)

HIGH-ENERGY LINES WITHIN CONTAINMENT

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.
54	Safety Depress System	Reactor Coolant Gas Vent Line from Branch-Off of Bleed Line to SD-1 to Juncture of RC-105/106 Header	Yes	Yes	5.1.2-1, 6.7-1
55	Safety Depress System	Reactor Vessel Vent Line to Orifice prior to RV-101 and RV-102	Yes	Yes	5.1.2-1, 6.7-1
56	CVCS	Letdown Line from Loop 2B to Regenerative Hx	Yes	Yes	9.3.4-1, 5.1.2-1
57	CVCS	Letdown Line from Regenerative Hx to Letdown Hx	Yes	Yes	9.3.4-1
58	CVCS	Letdown Line from Letdown Hx to Containment Pen	Yes	No	9.3.4-1
59	CVCS	Charging Line from Containment Pen to Regenerative Hx	Yes	No	9.3.4-1
60	CVCS	Charging Line from Regenerative Hx to RCS Loop 2A	Yes	Yes	9.3.4-1
61	CVCS	Auxiliary Spray Line to Pressurizer Spray Common Header	Yes	Yes	9.3.4-1
62	CVCS	SCS Hx Shutdown Purification Line Cont Pen Check Valve to Letdown Hx	Yes	Yes	9.3.4-1

TABLE 3.6-3 (Cont'd)**(Sheet 8 of 11)****HIGH-ENERGY LINES WITHIN CONTAINMENT**

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.
63	CVCS	RCP Seal Water Header from Cont Pent to Branch Seal Water Lines	Yes	No	9.3.4-1
64	CVCS	Letdown Line to Letdown Relief Valve Located Inside Cont between Inner and Outer CIVs	Yes	No	9.3.4-1
65	SIS	Safety Injection Line #1 (DVI) from Cont. Penetration to Check Valves SI-247	Yes	No	6.3.2-1B
66	SIS	Safety Injection Line #2 (DVI) from Cont. Penetration to Check Valve SI-246	Yes	No	6.3.2-1B
67	SIS	Safety Injection Line #3 (DVI) from Cont. Penetration Check Valve to Check Valve SI-245	Yes	No	6.3.2-1B
68	SIS	Safety Injection Line #4 (DVI) from Cont. Penetration Check Valve to Check Valve SI-244	Yes	No	6.3.2-1B
69	SIS	Safety Injection Line from SIS Tank #1 to DVI Line #1 Junction	Yes	No	6.3.2-1B

TABLE 3.6-3 (Cont'd)**(Sheet 9 of 11)****HIGH-ENERGY LINES WITHIN CONTAINMENT**

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.
70	SIS	Safety Injection Line from SIS Tank #2 to DVI Line #2 Junction	Yes	No	6.3.2-1B
71	SIS	Safety Injection Line from SIS Tank #3 to DVI Line #3 Junction	Yes	No	6.3.2-1B
72	SIS	Safety Injection Line from SIS Tank #4 to DVI Line #4 Junction	Yes	No	6.3.2-1B
73	SIS	SIS Tank #1 Relief Line to SIS Tank #1 Safety Valve	Yes	No	6.3.2-1B
74	SIS	SIS Tank #2 Relief Line to SIS Tank #2 Safety Valve	Yes	No	6.3.2-1B
75	SIS	SIS Tank #3 Relief Line to SIS Tank #3 Safety Valve	Yes	No	6.3.2-1B
76	SIS	SIS Tank #4 Relief Line to SIS Tank #4 Safety Valve	Yes	No	6.3.2-1B
77	SCS	Shutdown Cooling Return Line from Branch-Off to Check Valve SI-165 to Valve SD-673	Yes	Yes	6.3.2-1B

TABLE 3.6-3 (Cont'd)**(Sheet 10 of 11)****HIGH-ENERGY LINES WITHIN CONTAINMENT**

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.
78	SCS	Shutdown Cooling Return Line from Branch-Off to Check Valve SI-166 to Valve SD-672	Yes	Yes	6.3.2-1B
79	EFW	Emergency Feedwater Line from Motor Driven EFW Pump #1 Cont Pen Check Valve to SG #1 Common EFW Line	Yes	Yes	10.4.9-1 (Sheet 1)
80	EFW	Emergency Feedwater Line from Steam Driven EFW Pump #1 Cont Pen Check Valve to SG #1 Common EFW Line	Yes	Yes	10.4.9-1 (Sheet 1)
81	EFW	Emergency Feedwater Common Line to SG #1 Feedwater Downcomer	Yes	Yes	10.4.9-1 (Sheet 1)
82	EFW	Emergency Feedwater Line from Motor Driven EFW Pump #2 Cont Pen Check Valve to SG #2 Common EFW Line	Yes	Yes	10.4.9-1 (Sheet 1)
83	EFW	Emergency Feedwater Line from Steam Driven EFW Pump #2 Cont Pen Check Valve to SG #2 Common EFW Line	Yes	Yes	10.4.9-1 (Sheet 1)

TABLE 3.6-3 (Cont'd)**(Sheet 11 of 11)****HIGH-ENERGY LINES WITHIN CONTAINMENT**

<u>Item No.</u>	<u>System</u>	<u>Line Functional Description</u>	<u>Operating Pressure (>275 psig)</u>	<u>Operating Temperature (>200°F)</u>	<u>Figure No.</u>
84	EFW	Emergency Feedwater Common Line to SG #2 Feedwater Downcomer	Yes	Yes	10.4.9-1 (Sheet 1)

NOTE: See CESSAR-DC, Sections 3.6.1.1.1 and 3.6.1.1.2 for definitions of High- and Moderate-Energy Lines. The following systems are considered Moderate-Energy lines, since these systems do not operate during normal plant operation: Safety Injection System (SIS), Containment Spray System (CSS), Emergency Feedwater System (EFW) and Residual Heat Removal System (RHRS). Portions of these systems are, however, interfaced directly with other High-Energy systems or have sections which are High-Energy in nature. These sections have been included in this table.

I

TABLE 3.6-4

(Sheet 1 of 6)

HIGH-ENERGY LINES OUTSIDE CONTAINMENT (NOTE A)

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.	Notes
1	Main Steam	MSIV Bypass Line on LINE #2 to Bypass	Yes	Yes	10.1-2	B
2	Main Steam	MSIV Bypass Line on LINE #3 to Bypass Isolation Valve	Yes	Yes	10.1-2	B
3	Main Steam	Main Steam Atmospheric Dump Line on LINE #1	Yes	Yes	10.1-2	B
4	Main Steam	Main Steam Atmospheric Dump Line on LINE #2	Yes	Yes	10.1-2	B
5	Main Steam	Main Steam Atmospheric Dump Line on LINE #3	Yes	Yes	10.1-2	B
6	Main Steam	Main Steam Atmospheric Dump Line on LINE #4	Yes	Yes	10.1-2	B
7	Main Steam	Main Steam Safety Valve #1 on LINE #1	Yes	Yes	10.1-2	B
8	Main Steam	Main Steam Safety Valve #2 on LINE #1	Yes	Yes	10.1-2	B
9	Main Steam	Main Steam Safety Valve #3 on LINE #1	Yes	Yes	10.1-2	B
10	Main Steam	Main Steam Safety Valve #4 on LINE #1	Yes	Yes	10.1-2	B

I

TABLE 3.6-4 (Cont'd)**(Sheet 2 of 6)****HIGH-ENERGY LINES OUTSIDE CONTAINMENT (NOTE A)**

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.	Notes
11	Main Steam	Main Steam Safety Valve #5 on LINE #1	Yes	Yes	10.1-2	B
12	Main Steam	Main Steam Safety Valve #1 on LINE #2	Yes	Yes	10.1-2	B
13	Main Steam	Main Steam Safety Valve #2 on LINE #2	Yes	Yes	10.1-2	B
14	Main Steam	Main Steam Safety Valve #3 on LINE #2	Yes	Yes	10.1-2	B
15	Main Steam	Main Steam Safety Valve #4 on LINE #2	Yes	Yes	10.1-2	B
16	Main Steam	Main Steam Safety Valve #5 on LINE #2	Yes	Yes	10.1-2	B
17	Main Steam	Main Steam Safety Valve #1 on LINE #3	Yes	Yes	10.1-2	B
18	Main Steam	Main Steam Safety Valve #2 on LINE #3	Yes	Yes	10.1-2	B
19	Main Steam	Main Steam Safety Valve #3 on LINE #3	Yes	Yes	10.1-2	B
20	Main Steam	Main Steam Safety Valve #4 on LINE #3	Yes	Yes	10.1-2	B
21	Main Steam	Main Steam Safety Valve #5 on LINE #3	Yes	Yes	10.1-2	B
22	Main Steam	Main Steam Safety Valve #1 on LINE #4	Yes	Yes	10.1-2	B
23	Main Steam	Main Steam Safety Valve #2 on LINE #4	Yes	Yes	10.1-2	B

TABLE 3.6-4 (Cont'd)**(Sheet 3 of 6)****HIGH-ENERGY LINES OUTSIDE CONTAINMENT (NOTE A)**

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.	Notes
24	Main Steam	Main Steam Safety Valve #3 on LINE #4	Yes	Yes	10.1-2	B
25	Main Steam	Main Steam Safety Valve #4 on LINE #4	Yes	Yes	10.1-2	B
26	Main Steam	Main Steam Safety Valve #5 on LINE #4	Yes	Yes	10.1-2	B
27	Main Steam	Main Steam Line from LINE #1 to Emergency Feedwater Pump Turbine #1 Isolation Valve EF-108	Yes	Yes	10.1-2, 10.4.9-1 (Sheet 2)	B
28	Main Steam	Main Steam Line from LINE #4 to Emergency Feedwater Pump Turbine #2 Isolation Valve EF-109	Yes	Yes	10.1-2, 10.4.9-1 (Sheet 2)	B
29	Main Steam	EFW Pump Turbine #1 Drain Header to 4" x 1" Reducer	Yes	Yes	10.4.9-1 (Sheet 2)	
30	Main Steam	EFW Pump Turbine #2 Drain Header to 4" x 1" Reducer	Yes	Yes	10.4.9-1 (Sheet 2)	
31	Main Steam	Balance of Main Steam Piping on LINE #1	Yes	Yes	10.1-2	B
32	Main Steam	Balance of Main Steam Piping on LINE #2	Yes	Yes	10.1-2	B
33	Main Steam	Balance of Main Steam Piping on LINE #3	Yes	Yes	10.1-2	B

TABLE 3.6-4 (Cont'd)**(Sheet 4 of 6)****HIGH-ENERGY LINES OUTSIDE CONTAINMENT (NOTE A)**

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.	Notes
34	Main Steam	Balance of Main Steam Piping on LINE #4	Yes	Yes	10.1-2	B
35	SG Blowdown	SG #1 Common Blowdown Line from Containment Pen. to Pipe Exit of MSIV House	Yes	Yes	10.4.8-1	
36	SG Blowdown	SG #2 Common Blowdown Line from Containment Pen. to Pipe Exit of MSIV House	Yes	Yes	10.4.8-1	
37	Main Feedwater	SG #1 Main Feedwater Line	Yes	Yes	10.1-2	
38	Main Feedwater	SG #1 Downcomer Feedwater Line	Yes	Yes	10.1-2	
39	Main Feedwater	SG #2 Main Feedwater Line	Yes	Yes	10.1-2	
40	Main Feedwater	SG #2 Downcomer Feedwater Line	Yes	Yes	10.1-2	
41	CVCS	Letdown Line from Containment Pen to Main Letdown Orifice Including All Letdown Control Valve Lines	Yes	No	9.3.4-1 (Sheet 4)	
42	CVCS	Seal Injection Line from Charging Pumps to Seal Injection Hx	Yes	No	9.3.4-1 (Sheets 1,4)	

TABLE 3.6-4 (Cont'd)

(Sheet 5 of 6)

HIGH-ENERGY LINES OUTSIDE CONTAINMENT (NOTE A)

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.	Notes
43	CVCS	Seal Injection Line from Seal Injection Hx to Cont. Pen.	Yes	No	9.3.4-1 (Sheet 4)	
44	CVCS	Charging Pump #1 Line to Common Charging Line	Yes	No	9.3.4-1 (Sheet 1)	
45	CVCS	Charging Pump #2 Line to Common Charging Line	Yes	No	9.3.4-1 (Sheet 1)	
46	CVCS	Common Charging Line to Containment Penetration	Yes	No	9.3.4-1 (Sheets 1,4)	
47	CVCS	Charging Pump #1 Miniflow Line to Miniflow Orifice	Yes	No	9.3.4-1 (Sheet 1)	
48	CVCS	Charging Pump #2 Miniflow Line to Miniflow Orifice	Yes	No	9.3.4-1 (Sheet 1)	
49	CVCS	Charging Pump #1 Discharge to SIS DVI Lines. High-Energy to Isolation Valve CH-766	Yes	No	9.3.4-1 (Sheet 1)	

TABLE 3.6-4 (Cont'd)

(Sheet 6 of 6)

HIGH-ENERGY LINES OUTSIDE CONTAINMENT (NOTE A)

Item No.	System	Line Functional Description	Operating Pressure (>275 psig)	Operating Temperature (>200°F)	Figure No.	Notes
50	CVCS	Charging Pump #2 Discharge to SIS DVI Lines. High-Energy to Isolation Valve CH-707	Yes	No	9.3.4-1	

NOTES: A. See CESSAR-DC, Sections 3.6.1.1.1 and 3.6.1.1.2 for definitions of High- and Moderate-Energy Lines. The following systems are considered Moderate-Energy lines, since these systems do not operate during normal plant operation: Safety Injection System (SIS), Containment Spray System (CSS), Emergency Feedwater System (EFW) and Residual Heat Removal System (RHRS). Portions of these systems are, however, interfaced directly with other High-Energy systems or have sections which are High-Energy in nature. These sections have been included in this table. Portions of systems which continued into the Turbine Building were generally excluded from this table by the guidance of CESSAR-DC, Section 3.6.1.2.

B. The following nomenclature applies to Main Steam Lines:

LINE #1 refers to Main Steam Line from Steam Generator #1,
Line #1
LINE #2 refers to Main Steam Line from Steam Generator #1,
Line #2
LINE #3 refers to Main Steam Line from Steam Generator #2,
Line #1
LINE #4 refers to Main Steam Line from Steam Generator #2,
Line #2

Amendment E
December 30, 1988

VARIATION OF J-INTEGRAL
WITH LOADS FOR A TYPICAL CASE

Figure

3.6-1

APPENDIX 3.6A

**DISCUSSION OF FINITE DIFFERENCE ANALYSIS FOR
ANALYSIS OF PIPE WHIP**

E

EFFECTIVE PAGE LISTING

APPENDIX 3.6A

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E

1.0 EQUATIONS OF MOTION

The equations of motion used in the Finite Difference Analysis of Section 3.6.2.2.2 are of the form:

$$h (p_k - m_k y_k = -M_{k+1} + 2M_k - M_{k-1}) \quad (3.6A-1)$$

where:

- h = the node spacing
 p_k = the externally applied lateral loads at node k
 m_k = the lumped mass at node k
 y_k = the lateral deflection at node k
 M_k = the internal resisting moment in the beam at node k .

Power law moment-curvature relationship is assumed and the central difference approximation for the curvature,

$$\frac{1}{h^2} = (-y_{k+1} + 2y_k - y_{k-1}) \quad (3.6A-2)$$

is used.

A timewise central-difference scheme is used to solve the dynamic equations

$$y(t + \Delta t) = \Delta t^2 y(t) + 2y(t) - y(t - \Delta t) \quad (3.6A-3)$$

and for the first time step

$$y(t) = \Delta t^2 y(0) \quad (3.6A-4)$$

A time step equal to 1/10 the shortest period of vibration is used in the integration.

2.0 ELASTIC-PLASTIC MOMENT-CURVATURE LAW

The pipe is assumed to obey an elastic-strain hardening plastic moment-curvature law with isotropic strain hardening. The symbols used are defined as follows:

- M = moment
 $\bar{M}$ = current yield moment

E = elastic modulus of material at temperature
 I = moment of inertia
 Z = IE
 Φ = curvature
 Φ_c = M/Z = elastic curvature
 $\Delta\Phi_p$ = increment of plastic curvature
 Φ_p = $\epsilon |\Delta\Phi|$ = effective plastic curvature
 Φ_o = $\epsilon \Delta\Phi_p$ = permanent set curvature

At the end of each integration step, new values of Φ are calculated at each node.

The known values of Φ_p , Φ_o , and M at the start of the step are used to calculate M , $\bar{M}$ and $\Delta\Phi_p$ by the following procedure:

if $|\Phi - \Phi_o| < \bar{M}/Z$

$$M = Z (\Phi - \Phi_o) \quad (3.6A-5)$$

and

$$\Delta\Phi_p = 0 \quad (3.6A-6)$$

if $|\Phi - \Phi_o| > \bar{M}/Z$

$$M = \bar{M} = F(|\Phi - \Phi_o| + \Phi_p) \sin(\Phi - \Phi_o)$$

and

$$\Delta\Phi_p = \Phi - \Phi_o - M/Z$$

where

$$F(\Phi) = K(\Phi)^n.$$

3.0 POWER LAW MOMENT-CURVATURE RELATIONSHIP

The following stress strain law is assumed in the plastic range:

$$\sigma = K(\epsilon)^n \quad (3.6A-7)$$

The corresponding moment-curvature law is:

$$M = K(\Phi)^n \quad (3.6A-8)$$

where:

$$K = \frac{2}{3+n} \Pi (R_o^{3+n} - R_i^{3+n}) \frac{\Gamma[(1/2)n + 1]}{\Gamma[(1/2)n + 3/2]} \bar{K} \quad (3.6A-9)$$

or, to a good approximation,

$$K = \frac{4\bar{K}}{3+n} (1 - .291n - .076n^2) (R_o^{3+n} - R_i^{3+n}) \quad (3.6A-10)$$

in which:

R_o = pipe outside radius

R_i = pipe inside radius

In the elastic range the moment-curvature law is:

$$M = EI\phi \quad (3.6A-11)$$

The transition from elastic to plastic behavior on initial loading occurs at:

$$\phi = \frac{\frac{1}{n-1}}{(EI) \frac{1}{K}} \quad (3.6A-12)$$

4.0 STRAIN RATE EFFECTS

The effect of strain rate in carbon steel is accounted for by using a rate dependent stress strain law of the form:

$$\sigma(\epsilon, \dot{\epsilon}) = \left\{ 1 + \frac{\dot{\epsilon}}{(40.4)} \right\}^{1/5} G(\epsilon) \quad (3.6A-13)$$

Where $G(\epsilon)$ is the static stress strain relationship. For stainless steel, the effect of strain rate is less pronounced so that a 10% increase in yield and ultimate strength is used.

5.0 RESTRAINT BEHAVIOR

The analysis is capable of handling a bilinear or power law restraint behavior. The behavior of the restraint is unidirectional. The restraint unloads elastically only to zero state, being left with a permanent set, and reloads along a bilinear or power law curve.

3.7 SEISMIC DESIGN

3.7.1 SEISMIC INPUT

3.7.1.1 Seismic Input

This section discusses the seismic design parameters and methodologies being used for the design of those systems and subsystems important to safety and classified as Category I in Section 3.2.

The System 80+ Standard Design as defined by CESSAR-DC is not based on a specific site. Generic site conditions were selected to cover a range of possible conditions for the System 80+ sites. More specifically, sets of representative cases from each of four generic site categories were evaluated to create the ground surface and foundation level spectra shown in Figures 3.7-1 through 3.7-24. Out of 12 soil cases analyzed in Section 2.5.2, nine are used in the soil structure interaction (SSI) analysis. The three cases eliminated in the SSI analysis (A1, B3 and D1) were non-governing cases whose soil response levels were enveloped by other cases. See Section 2.5.2 for details of this analysis phase.

The effect of differential seismic displacement on the equipment and supports is included in the analysis as described in Section 3.7.2.1. I

3.7.1.2 Design Time History

For the time history method of analysis, three design time histories are generated that are consistent with the design rock outcrop spectra at the free field. The characteristics of each time history are presented in Section 2.5.2.5.1. The response spectra plots for these time histories are shown in Figures 3.7-25 through 3.7-27.

3.7.1.3 Critical Damping Values

Damping values used for various nuclear safety-related structures systems and components are based upon Regulatory Guide 1.61 or ASME Code Case N-411-1 (See Figure 3.7-41). These values are expressed in percent of critical damping and are given in Table 3.7-1. When the response spectra method of analysis is used for piping, damping values are based on Code Case N-411-1.

3.7.1.4   Supporting Media for Seismic Category I Structures

Category I structures are founded directly on rock or competent soil. The foundation embedment depth for System 80+ standard plant is 52 feet (Reference 21). The rock properties and the layering characteristics, including shear wave velocity, shear modulus, and density, are given in Section 2.5. The System 80+ Standard Plant is designed for the range of soil conditions discussed in Section 2.5 and shown in Appendix 3.7B.

3.7.1.4.1   Soil Structure Interaction (SSI)

Two different types of analysis methodologies are used for the seismic analyses. For the fixed-base cases, modal superposition time history analyses are performed using the rock outcrop motions as control motions. When a structure is supported on soil, the SSI is taken into account by coupling the structural model with the soil medium. To accomplish this, the methodology of the computer program SASSI (System for Analysis of Soil Structure Interaction, Reference 6) is used. Detailed methodology and results of the SSI analysis are presented in Appendix 3.7B.

3.7.2   SEISMIC SYSTEM ANALYSIS

3.7.2.1   Seismic Analysis Method

3.7.2.1.1   Seismic Category I Structures, Systems, and Components Other Than NSSS

The Reactor Building (RB) is modeled as a multi-degree of freedom system for the seismic analysis. Figures 3.7-28 through 3.7-30 show typical sketches of the three structural components of the overall model - Internal Structure (IS), Shield Building (SB), and Steel Containment Vessel (SCV). Figure 3.7-31 is a schematic representation of the combined structural model of the RB. The RB is modeled as a lumped mass-spring model.

Further details of dynamic modeling of building structures for seismic analysis are described in Section 3.7.2.3. The horizontal models are analyzed for the plant E-W direction and N-S direction excitations and the vertical model for vertical excitation. The results are then combined as described in Section 3.7.2.6. The seismic analysis of the above systems is performed by one of the following methods:

3.7.2.1.1.1 Response Spectrum Method of Analysis

The response of a multi-degree-of-freedom system subjected to seismic excitation is represented by the following differential equation of motion:

$$[M] [\ddot{\{X\}} + \{\ddot{U}_g\}] + [C] \{\dot{X}\} + [K] \{X\} = 0$$

where:

$[M]$ = mass matrix (n x n)

$[C]$ = damping matrix (n x n)

$[K]$ = stiffness matrix (n x n)

$\{X\}$ = column vector of relative displacements (n x 1)

$\{\dot{X}\}$ = column vector of relative velocities (n x 1)

$\{\ddot{X}\}$ = column vector of relative accelerations (n x 1)

n = number of dynamic degrees of freedom

$\{\ddot{U}_g\}$ = column vector of ground accelerations (n x 1)

In the response spectrum method of analysis, the equations of motion are decoupled using the transformation:

$$\{X\} = [\phi] \{Y\}$$

where:

$[\phi]$ = mode shape matrix

$\{Y\}$ = vector of normal, or generalized coordinates (m x 1)

m = number of modes considered

The decoupled equation of motion for each mode is transformed to a single degree of freedom system:

$$\ddot{Y}_j + 2 \lambda_j \omega_j \dot{Y}_j + \omega_j^2 Y_j = -\Gamma_j \ddot{U}_g$$

where:

Y_j = generalized coordinate of j^{th} mode

λ_j = damping ratio for the j^{th} mode expressed as fraction of critical damping

ω_j = circular frequency of j^{th} mode of the system

r_j = modal participation factor of the j^{th} mode

$$= \frac{\{\phi_j\}^T [M] \{1\}}{\{\phi_j\}^T [M] \{\phi_j\}}$$

The generalized maximum response of each mode is determined from:

$$Y_j(\text{max}) = r_j \frac{S_{aj}}{\omega_j^2}$$

where S_{aj} is the spectral acceleration corresponding to frequency ω_j .

The maximum displacement of node i relative to the base due to mode j is:

$$X_{ij}(\text{max}) = \phi_{ij} Y_j(\text{max})$$

The modal response $X_{ij}(\text{max})$ is used to determine other modal response quantities, such as forces. As the maximum modal response do not occur at the same time, modal superposition is done to obtain the final response by the double sum method described in Section 3.7.2.7.

3.7.2.1.1.2 Time History Method

The solution of the differential equation of motion given in Section 3.7.2.1.1.1 can be obtained by the method of modal superposition or by the method of direct integration.

A. Modal Superposition Method

The modal superposition method is used when the equations of motion can be decoupled as given in Section 3.7.2.1.1.1. Then the decoupled equation of motion for each mode is integrated using a proven technique, and the total response is obtained by superposition method.

B. Direct Integration Method

In this method, direct integration of the equations of motion by either implicit or explicit methods of numerical integration are used to solve the equations of motion.

For commonly used implicit methods, ΔT is not larger than 1/10 of the shortest period of interest.

For explicit methods, the time step is also a function of the element size used in the model and is established on the basis of element size to ensure stability of the response.

3.7.2.1.2 Seismic Analysis Method for the NSSS**3.7.2.1.2.1 Introduction**

The major components of the reactor coolant system are designed to the appropriate stress and deformation criteria of ASME Code, Section III, for the set of loadings included in the component design specification. The adequacy of seismic loadings used for the design of the major components of the reactor coolant system are confirmed by the methods of dynamic analysis employing time history and response spectrum techniques. The major components are the reactor vessel, the steam generators, the reactor coolant pumps, the reactor coolant main loop piping, the surge line and the pressurizer.

Detailed dynamic models of the building structures and the NSSS are generated. Based on these detailed models, equivalent, simplified dynamic models are developed. The simplified building and NSSS models are combined and translated into a form suitable for input to the SSI analysis code (see Section 3.7.1.4.1). A number of soil cases are modeled and the time history analyses are performed. The soil cases are chosen to envelope all potential building sites. The results of these analyses are contained in Appendix 3.7B. These results, the simplified building model(s), and the detailed NSSS model are used to perform the analysis discussed in Section 3.7.2.1.2.3.

A composite three-dimensional lumped-mass model of the reactor vessel, the two steam generators, the four reactor coolant pumps, the pressurizer, and the interconnecting main loop piping is coupled with a three-dimensional lumped-mass model of the reactor building for performing the analysis of these dynamically coupled components of the reactor coolant system. In addition, the representation of the reactor vessel assembly used in this coupled model includes sufficient detail of the reactor internals to account for possible dynamic interaction between the reactor coolant system and internals. The seismic input excitation is the basemat acceleration time histories. The results of this

analysis include appropriate excitation data for use in separate analyses of the surge line and of a more detailed model of the reactor internals. A model of the coupled components of the reactor coolant system is shown in Figure 3.7-32. A model of the pressurizer is shown in Figure 3.7-33.

The analysis of the surge line piping employs a separate mathematical model and utilizes either response spectrum or time-history techniques.

For the time history analyses, dynamic responses are obtained by applying the vertical and two orthogonal horizontal seismic excitations simultaneously to the coupled model.

The square root of the sum of the squares (SRSS) methods is normally used to combine the modal responses when the response spectrum modal analysis method is employed. In those cases, however, where modal frequencies are closely spaced, the responses of the closely spaced modes are combined by the sum of the absolute values method and, in turn, combined with the responses of the remaining significant modes by the square root of the sum of the square method, as described in Section 3.7.2.7. Contributions from all significant modes of response are retained in the analyses.

The damping factors used in analysis of Seismic Category I structures, systems and equipment are selected from Table 3.7-1. Because the coupled reactor coolant system building interior structure model represents steel and concrete structures, composite modal damping consistent with Table 3.7-1 is used for the SSE seismic analysis. The damping factors given in Table 3.7-1 include those recommended in Regulatory Guide 1.61. The results of the dynamic analyses of the major components of the reactor coolant system, which are performed to confirm the adequacy of the seismic design, are contained in Appendix 3.7A.

3.7.2.1.2.2 Mathematical Models

In the descriptions of the mathematical models that follow, the spatial orientations are defined by the set of orthogonal axes for which Z represents the vertical direction and X and Y are in the horizontal plane in the directions indicated on the appropriate figure. The mathematical representation of the section properties of the structural elements employs a 12 x 12 stiffness matrix for the three-dimensional space frame models, and employs a 6 x 6 stiffness matrix for the two-dimensional plane frame model. Elbows in piping runs include the in-plane/out-of-plane bending flexibility factors as specified in the ASME Code, Section III.

A schematic diagram of the composite mathematical model used in the analyses of the dynamically coupled components of the reactor coolant system is presented in Figures 3.7-32 and 3.7-33. This model includes 36 mass points with a total of 96 dynamic degrees of freedom to represent the RCS, including the pressurizer. Additional mass points and dynamic degrees of freedom, not shown in the figures, are used to represent the containment building and interior structures in the coupled seismic model. The surge line is very flexible relative to the rest of the structure, and is not considered in the coupled model analysis. The pressurizer is mathematically coupled to the remainder of the RCS by way of the building structure represented in the coupled seismic model. The mass points and corresponding dynamic degrees of freedom are distributed to provide appropriate representations of the dynamic characteristics of the components, as follows:

- A. The reactor vessel, with internals, is represented by 4 mass points with a total of 11 dynamic degrees of freedom.
- B. Each of the two steam generators is represented by 4 mass points with a total of 10 dynamic degrees of freedom, each of the four reactor coolant pumps is represented by 2 points with a total of 6 dynamic degrees of freedom.
- C. The pressurizer is represented by 6 mass points with a total of 13 dynamic degrees of freedom; each branch of cold leg piping is represented by a mass point with 3 dynamic degrees of freedom.
- D. Each branch of hot leg piping is represented by a single mass point with 2 dynamic degrees of freedom. The representation of the reactor vessel internals is formulated in conjunction with the analysis of the reactor vessel internals discussed in Section 3.7.3.14, and is designed to simulate the dynamic characteristics of the models used in that analysis.

The mathematical model provides a three dimensional representation of the dynamic response of the coupled components to seismic excitations in both the horizontal and vertical directions. The mass is distributed at the selected mass points and corresponding translational degrees of freedom are retained to include rotary inertial effects of the components. The total mass of the entire coupled system is dynamically active in each of the three coordinate directions.

Surge Line

A lumped parameter, multimass mathematical model is employed in the analysis of the surge line. The model is shown schematically in Figure 3.7-34. The surge line is modeled as a

three-dimensional piping run with end points anchored at the attachments to the pressurizer and the reactor vessel outlet piping. All supports defined for the surge line assembly are included in the mathematical model. The total mass of the surge line is dynamically active in each of the three coordinate directions. The surge line is analyzed as uncoupled from the reactor coolant system, using the motions of the hot leg, pressurizer and supports as input.

3.7.2.1.2.3 Analysis

Modeling and analysis of the coupled components of the reactor coolant system and the pressurizer are performed using MDC STRUDL. A description of MDC STRUDL is given in Section 3.9.1.2.1.1. Modeling and analysis of the surge line is performed using the SUPERPIPE code, a description of which is given in Section 3.9.1.2.1.4.

Time history data for all six possible components of motion are applied simultaneously to the coupled building model to analyze the coupled components of the reactor coolant system.

The responses to seismic excitation for the coupled components of the reactor coolant system are computed using the transient analysis capability of MDC STRUDL. The response of the surge line is computed using the spectrum or transient analysis capability of SUPERPIPE. Input excitations at the endpoints for the surge line analysis are generated by MDC STRUDL using the results of the coupled containment building interior structure, reactor coolant system analysis. The input excitations at the intermediate supports are developed from the building analyses.

The reaction forces and moments obtained from the dynamic seismic analysis are compared with seismic loads in each component design specification. The results of this comparison are summarized in tabular form in Appendix 3.7A for the locations of maximum calculated load.

Since the three directions of earthquake motion are statistically independent, the maximum responses are calculated by a simultaneous application of motion resulting from all three directions of earthquake.

The maximum seismic loads calculated by the response spectrum techniques are the result of combining the modal reactions due to both horizontal and vertical excitations. The method of modal combination is discussed in Section 3.7.2.7. The maximum responses due to each of the three earthquake components are then combined by the SRSS method.

The seismic loadings specified for the design of the reactor coolant system components and supports are adequate because all seismic loads calculated by the dynamic seismic analyses are less than the corresponding loads in the component design specifications.

3.7.2.2   Natural Frequencies and Response Loads

These data are provided in Appendix 3.7A.

3.7.2.3   Procedures Used for Analytical Modeling

3.7.2.3.1   Modeling of the NSSS and BOP

The procedure used for modeling NSSS components and interconnecting piping is described in Section 3.7.2.1.2. The procedure used for modeling BOP components and interconnecting piping is described in the following sections.

3.7.2.3.2   Designation of Systems Versus Subsystems

The calculation of the dynamic response of a nuclear power plant subject to an earthquake loading is divided into two categories. The first is the "safety-related main structural system" and the second is the "safety-related subsystem". The "safety-related main structural system" category refers to the analysis of major buildings and structures which house and/or support safety-related systems. The "safety-related subsystems" category refers to smaller safety-related structures, systems, and components.

The major structures which are analyzed in the main structural system analysis are:

- A. Steel containment vessel, internal structure and shield building, and
- B. Nuclear Annex, and turbine building

3.7.2.3.3   Decoupling Criteria for Subsystems

In general, all subsystems such as equipment and piping with the exception of the reactor coolant system are decoupled from the floor which supports them. As recommended in the Standard Review Plan, Section 3.7.2, the following decoupling criteria is used in instances where a subsystem needs to be modeled:

- A. If $R_m < 0.01$, decoupling is done for any R_f .
- B. If $0.01 < R_m < 0.1$, decoupling is done if $R_f \leq 0.8$ or $R_f \geq 1.25$

- C. If $R_m > 0.1$, an approximate dynamic model of the subsystem is included in the main structural system.

where:

$$R_m = \frac{\text{total mass of the supported subsystem}}{\text{total mass of the supporting system}}$$

$$R_f = \frac{\text{fundamental frequency of the supported subsystem}}{\text{dominant frequency of the support motion}}$$

The masses of the decoupled subsystems are included with the structural mass of the supporting floor slabs in the system model. The containment internal structure model includes a simplified dynamic model of the reactor coolant system.

3.7.2.3.4 Lumped Mass Considerations

The safety-related structures are modeled as a multi-degree of freedom system. The major structural element systems such as floor slabs, foundation mat, roof slab, shear walls and braced vertical frames are included in the model. All subsystems such as equipment and piping are considered in accordance with the decoupling criteria described in Section 3.7.2.3.3.

The Shield Building and Steel Containment Vessel are each analyzed using a single model. Two independent models are used for the seismic analysis of the Internal Structure. One model is used for the horizontal excitations and the other is used for the vertical excitation. Different models are used in order to better represent the structural behavior for the different excitations. Each of the models and the method of lumping the masses are described below.

3.7.2.3.4.1 Model for Horizontal Excitation

The Power Generation Complex (PGC) of the System 80+ consists of the following structures:

- o Reactor Building
- o Nuclear Annex

The modeling approach that is used for the Reactor Building (RB) structural model consists of developing a 3-D finite element model (FEM) of the Internal Structure (IS) and an axisymmetric FEM of the Shield Building (SB) and, based on the FEM models, developing equivalent 3-D lumped parameter stick models. This approach is used for the IS and the SB. Because of its

slenderness, the Steel Containment Vessel (SCV) has significant "membrane-type" action when it vibrates, and it is explicitly modeled with shell elements.

3.7.2.3.4.1.1 Development of FEM and Stick Models of the Internal Structure

The FEM of the IS is developed by defining major floor elevations and major elevations at which significant stiffness discontinuities occur across the entire area of the structure. Twelve such elevations were selected, as follows:

+50.00 ft.	Top of Basemat
+63.50 ft.	Second Floor
+79.50 ft.	Reactor Vessel Column Supports
+90.25 ft.	Third Floor (at Ground Level)
+104.50 ft.	Steam Generator Supports
+114.00 ft.	Fourth Floor
+120.00 ft.	Top of Reactor Vessel
+144.50 ft.	Operating Floor
+164.33 ft.	Main Steam Line Supports
+178.00 ft.	Top of Steam Generator Shield Walls (stiffness discontinuity)
+191.33 ft.	
+210.00 ft.	Top of Crane Wall

The load-resisting elements of each floor consist of concrete walls. These walls are modeled with quadrilateral shell elements or solid 8-node elements depending on the thickness of the walls. Concrete slabs of significant thickness are modeled with quadrilateral shell elements. For all seismic analyses, only dead load is assumed to contribute to the inertial forces of the IS. No live load is included in the computation of floor masses since live loads in the RB are minimal compared to dead loads. The translational mass and mass moments of inertia are lumped at the center of mass of each floor. This is done for ease of comparison between the full 3-D FEM and the equivalent 3-D stick model. The mass of each floor includes the mass of concrete walls, concrete slabs, concrete columns, heavy steel platforms, and heavy equipment. For light equipment, secondary structural steel, piping, tanks and miscellaneous mechanical and electrical components, a cumulative uniformly distributed mass is estimated and added to each floor.

Figure 3.7-28 shows a schematic of the stick model of the IS for the horizontal analysis. To account for torsional and rocking response at the outer edges of the IS, massless rigid links are connected to the center-of-mass of each major elevation of the stick model and are extended to the four edges of the structure at each floor elevation. For the design of in-structure components that are located closer to the edges of each floor, the time history response at the outer end of these links is used to generate response spectra.

3.7.2.3.4.1.2   Development of FEM and Stick Models of the Shield Building

Since the SB is symmetric about the vertical axis of the RB, the FEM of the SB is developed using an assembly of axisymmetric shell elements. Fixed-base modal analyses are performed for the horizontal and vertical directions and, based on these analyses, mass and stiffness properties are selected for the SB stick model. The mass of the SB is lumped at eleven nodal points along the height of the stick.

To account for torsional and rocking response at the edge of the SB, massless rigid links are connected at the center-of-mass of each elevation and are extended to the edge, as shown in Figure 3.7-29, similar to the IS stick model.

3.7.2.3.4.1.3   FEM of Steel Containment Vessel

The SCV is modeled with shell elements as shown in Figure 3.7-30. The bottom nodes, corresponding to elevation +91 ft., are connected with rigid links to the stick model of the IS.

3.7.2.3.4.1.4   Combined Model of Power Generation Complex Structures

The RB model is constructed by connecting the stick models of the IS, the SB, the NSSS and the FEM of the SCV, as shown in Figure 3.7-31. The IS and SB stick models are essentially co-axial (except for floor eccentricities in the IS). The IS and the SB stick models are connected with a rigid link at elevation +115 ft. The SCV is also connected to the IS with rigid links at elevation +91 ft. Also shown in Figure 3.7-31 are the stick models of the adjacent structures, which are not connected to the RB above the foundation.

3.7.2.3.4.2   Model for Vertical Excitation

The previous discussion of the models developed for horizontal excitation applies to vertical excitation model development, with a minor change in the case of the IS model. The only difference between the horizontal and vertical analysis stick models for the IS is the eccentricity of the center of mass to the center of rigidity at each major elevation.

3.7.2.3.5   Modeling for Three Component Input Motions

As discussed in Section 3.7.2.3.4, two independent models, one in the horizontal and the other in the vertical direction, are used. The horizontal and vertical models are decoupled, since the response in the vertical direction due to horizontal excitation will be negligible and vice versa. In the horizontal analysis of

an asymmetrical structure, the seismic model is analyzed along the plant E-W direction and N-S direction. For symmetrical structures the model is analyzed along any one principal axis, since the response along both the principal axes is the same.

3.7.2.4   Soil Structure Interaction (SSI)

The soil model and SSI analysis methodologies are described in Appendix 3.7B.

3.7.2.5   Development of Floor Response Spectra

The time history method of analysis is used to generate the floor response spectra. The spectra are generated according to the procedure given in Regulatory Guide 1.122. As discussed in Section 3.7.2.3.4, the horizontal and vertical models are decoupled and the floor response in horizontal and vertical directions are obtained by three separate analyses. For horizontal analysis, the response spectra are generated for each floor along the two axes of the structure. In vertical analysis, the response spectra are generated for the slabs.

The spectra are generated for appropriate critical damping for OBE and SSE. The peaks of the response spectra are broadened as described in Section 3.7.2.9.

3.7.2.6   Three Components of Earthquake Motion

3.7.2.6.1   Seismic Category I Structures, Systems, and Components Other Than NSSS

The three independent orthogonal components of earthquake motion (2 horizontal and 1 vertical) are applied to the structural models as separate loading cases. The models are analyzed using either the time-history or response spectrum method of dynamic analysis as appropriate. The total response of the structure due to the three input seismic motions is obtained by combining the directional responses using the square root sum of the squares (SRSS) method.

3.7.2.6.2   Nuclear Steam Supply System

The procedures for considering the effects of three components of earthquake motion in determining the seismic response of NSSS systems, components and supports are in accordance with Regulatory Guide 1.92. They are discussed in Section 3.7.2.1.2.3.

3.7.2.7   Combination of Modal Responses**3.7.2.7.1   Seismic Category I Structures, Systems, and
Components Other Than NSSS**

The total seismic response of a structure to an input response spectrum loading is obtained by combining the response of each individual mode of the structure in accordance with the requirements of Regulatory Guide 1.92. If the modes are not closely spaced (i.e. no two consecutive modes have frequencies which differ from each other by 10 percent or less) then the significant modes are combined using the square root sum of the squares (SRSS) of the corresponding maximum values of the response of each element of the structure. This is expressed mathematically as:

$$R = \left(\sum_{k=1}^N R_K^2 \right)^{1/2}$$

Where R is the maximum response of a given element, R_K is the peak response of the element due to the K^{th} mode, and N is the number of significant modes.

If some of the modes are closely spaced the response of the individual modes is combined using the Ten Percent Method from Regulatory Guide 1.92. This can be expressed as:

$$R = \left(\sum_{k=1}^N R_K^2 + 2 \sum_{i \neq j} |R_i R_j| \right)^{1/2}$$

Where R, R_K and N are as previously defined. The second summation is performed on all i and j modes whose frequencies are closely spaced to one another. Alternative summation methods given in Regulatory Guide 1.92, such as the Double Sum Method, are acceptable substitutes for the method described above.

3.7.2.7.2   Nuclear Steam Supply System

The SRSS method is the procedure normally used to combine the modal responses when the modal analysis response spectrum method of analysis is employed. The procedure, in accordance with Regulatory Guide 1.92, is modified in two cases:

- A. In the analysis of simple system where three or less dynamic degrees of freedom are involved, the modal responses are combined by the summation of the absolute values method;

- B. In the analysis of complex system where closely spaced modal frequencies are encountered, the responses of the closely space modes are combined by the summation of the absolute values method and, in turn, combined with the responses of the remaining significant modes by the SRSS method. Modal frequencies are considered closely spaced when their difference is less than ± 10 percent of the lower frequency.

3.7.2.8   Interaction of Non-Safety-Related Structures with Safety-Related Structures

When safety-related and non-safety-related structures are integrally connected, the non-safety-related structure is included in the model when determining the forces on safety-related structures. Such non-safety-related structures (as well as non-safety-related structures adjacent to safety-related structures) are designed so that their failure under SSE conditions will not cause the failure of the safety-related structures.

3.7.2.9   Effects of Parameter Variations on Floor Response Spectra

To account for the expected variation in structural properties, dampings and other parameter variations, the peaks of floor response spectrum curves are broadened by $\pm 15\%$ and smoothed in accordance with Regulatory Guide 1.122.

Soil property related spectrum peaks are further broadened, where required, to conservatively account for all potential variations of soil properties within the envelope of site conditions. Structures, systems and equipment are qualified to either 1) the envelope of the collective broadened spectra for all soil cases comprising the site envelope or 2) the broadened spectra for each of the soil cases which comprise the site envelope.

3.7.2.10   Use of Constant Vertical Static Factors

A constant seismic vertical load factor is not used for the seismic design of Seismic Category I structures, systems, components and equipment.

The safety-related structures, systems, and components are analyzed in the vertical direction using the methods described in Section 3.7.2.1. Based on the vertical seismic analysis, a vertical static factor is determined to design columns and shear walls. The vertical floor flexibilities are accounted for in the response spectra at each individual floor elevation of the building structures. The floor beams are designed statically for the acceleration value obtained per Reference 1.

3.7.2.11   Methods Used To Account for Torsional Effects

The mathematical models used in analysis of Seismic Category I systems, components, and piping systems include sufficient mass points and corresponding dynamic degrees-of-freedom to provide a three-dimensional representation of the dynamic characteristics of the system. The distribution of mass and the selected location of mass points account for torsional effects of valves and other eccentric masses.

The structural models used for Seismic Category I systems are constructed with elements containing 6 degrees of freedom per node, incorporating torsional effects into the models. Torsional effects are also accounted for in the building models used to generate floor response spectra. An additional eccentricity of 5% of the maximum building dimension was included to account for accidental torsion.

3.7.2.12   Comparison of Responses

With the exception of the surge line, the time-history method is used for structural analysis of the NSSS and the associated building structures. Therefore, responses obtained from the response spectrum and time-history methods are not compared.

3.7.2.13   Methods for Seismic Analysis of Dams

If applicable for the site, analyses of safety-related dams will be performed taking into account appropriate factors such as the behavior of dam material under horizontal and vertical seismic loadings. Dam soil-structure-interaction effects will be considered.

**3.7.2.14   Determination of Safety-Related Structure
Overturning Moments**

The overturning moments and base shears due to seismic forces for Category I structures are determined using the response spectrum method of analysis. The seismic motion is input to the structural models in three independent orthogonal directions. The overturning moments for shell structures are automatically included in the analysis of this type of structure.

3.7.2.15   Analysis Procedure for Damping

For modal superposition method, composite modal damping values are used for structures with components of different damping characteristics. The composite modal damping values are based on weighting the damping factors according to the mass or the stiffness of each element. For the mass proportional damping, formulation is as follows:

$$\beta_j = \frac{\sum_{i=1}^N \{\phi_j\} \beta_i [M_i] \{\phi_j\}}{\{\phi_j\}^T [M] \{\phi_j\}}$$

where:

- n = total number of components,
- β_j = composite modal damping for mode j ,
- β_i = critical modal damping associated with component i ,
- ϕ_j = mode shape vector,
- $\{M_i\}$ = subregion of mass matrix associated with component i , and
- $[M]$ = the mass matrix of the system.

For direct integration method, viscous damping proportional to the mass and stiffness matrix is used; thus

$$[C] = \alpha [K] + \beta [M]$$

where $[C]$ is the damping matrix, $[K]$ is the stiffness matrix and $[M]$ is the mass matrix. The values of α and β are selected such that the damping in the range of frequency of interest is approximately equal to the damping of the structure.

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

3.7.3.1 Seismic Analysis Methods

The seismic analysis of the Seismic Category I structures, subsystems, and components other than piping is performed by either the response spectrum or time history method as described in Section 3.7.2.1.1 or an equivalent static method described in Section 3.7.3.5.

For Seismic Category I piping, each piping system is idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping subsystem is determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as changes in stiffness due to curved

members. Generally, a uniform response spectrum analysis is performed using the envelope of all applicable spectra to account for inertia effects. The total seismic response of the piping is then calculated by adding the results of the response spectrum analysis to a static analysis which accounts for the relative displacement effects between support locations. Since the displacement effects are self-limiting, it is justified to place them in the secondary stress category.

As an alternative to the modal response method, a time history method of analysis may be used. This method is also used for other types of dynamic analyses such as LOCA and hydraulic transients. Either a direct integration method or a modal superposition method is used to solve the equations of motion.

3.7.3.2   Determination of Number of Earthquake Cycles

The procedure used to account for the fatigue effect of cyclic motion associated with the OBE recognizes that the actual motion experienced during a seismic event consists of a single maximum or peak motion, and some number of cycles of lesser magnitude. The total or cumulative usage factor can also be specified in terms of a finite number of cycles of the maximum or peak motion. Based on this consideration, Seismic Category I subsystems, components, and equipment are designed for a total of 200 full-load cycles, about a mean value of zero, and with an amplitude equal to the maximum response produced during the entire OBE event.

3.7.3.3   Procedure Used for Modeling

The modeling techniques incorporate either a single or multidegree of freedom subsystem consisting of discrete masses connected by spring elements. The associated damping coefficients are consistent with Table 3.7-1. The degree of complexity of each model is sufficient to accurately evaluate the dynamic behavior of the component. For additional details on pipe modeling, see the section below.

Valves (i.e., with natural frequencies greater than the frequency corresponding to the zero period acceleration (ZPA)) are included in the piping system model as lumped masses on rigid extended structures. If it is shown by test or analysis that a valve has a frequency less than a frequency corresponding to the ZPA, then a multimass, dynamic model of the valve, including the appropriate stiffnesses, is developed for use in the piping system model.

The continuous piping system is modeled as an assemblage of beams. The mass of each beam is lumped equally at its associated end nodes, which are connected by massless elastic members, representing the physical properties of each segment. The pipe lengths between these mass points shall not be greater than the length that would produce a natural frequency equal to the cutoff frequency (ZPA) when calculated based on a simply supported beam. All concentrated weights on the piping system such as main valves, relief valves, pumps, and motors are modeled as lumped masses. The torsional effects of the valve operators and other equipment with offset center of gravity with respect to centerline of the pipe are included in the analytical model.

3.7.3.4   Basis for Selection of Frequencies

The basis for acceptability of the seismic design of equipment and subsystems is that the stresses and deformations produced by vibratory motion of the postulated seismic events, in combination with other coincident loadings, be within the established limits.

Within practical limitations, the seismic design is accomplished in a manner to maintain the resonant frequencies outside the range that is significantly excited by the forcing frequencies. The stiffness of the restraint and supports system is designed to maintain the fundamental frequencies of equipment and subsystems sufficiently removed from the resonant range and, thereby, maintain the seismic response within the established limits. If the natural frequencies of the equipment and supporting structures are in the same range where resonance can occur, the resonance is accounted for in the analysis.

3.7.3.5   Use of Equivalent Static Load Method of Analysis

The equivalent static load method involves the multiplication of the total weight of the equipment or component member by the specific seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient is established on the basis of the expected dynamic response characteristics of the component. Components that can be adequately characterized as a single degree of freedom system are considered to have a modal participation factor of one. Seismic acceleration coefficients for multi-degree of freedom systems which may be in the resonance region of the amplified response spectra curves are increased by 50% to account conservatively for the increased modal participation. If the equipment natural frequency is above the frequency corresponding to the zero period acceleration (ZPA), the seismic acceleration coefficient is equal to 1.0 times the ZPA.

3.7.3.6   Three Components of Earthquake Motion

Seismic responses resulting from analysis of subsystems due to three components of earthquake motions are combined in the same manner as the seismic response resulting from the analysis of building structures (Section 3.7.2.6).

The following description is applicable to safety-related components and systems:

The system and equipment response is determined using three earthquake components, two horizontal and one vertical. Floor response spectra are generated for two perpendicular horizontal directions, (i.e., N-S, E-W) and the vertical direction. Piping and equipment analysis is performed with these response spectra components applied in the N-S, E-W, and vertical directions.

The damping values used in the analysis of equipment are those given in Table 3.7-1.

3.7.3.7   Combination of Modal Responses

When a response spectrum method of analysis is used to analyze a subsystem, the maximum response (accelerations, shears, and moments) in each mode is calculated independent of time. If the frequencies of the modes are well separated, the SRSS method of mode combination gives acceptable results; however, where the structural periods are not well separated, the modes are combined in accordance with Regulatory Guide 1.92. This is automatically performed by the piping analysis computer program SUPERPIPE.

The effects of seismic response of supports and equipment are not directly included in the seismic analysis of piping initially as equipment and supports are normally designed and analyzed subsequent to the piping analysis.

3.7.3.8   Analytical Procedures for Piping

All Seismic Category I piping is analyzed for seismic effects as described in Section 3.7.3.1.

3.7.3.8.1   Dynamic Analysis

Each piping system is idealized as a mathematical model consisting of lumped masses connected by elastic massless members. Appendages having significant dynamic effects on the piping system, such as motors attached to motor-operated valves, are included in the model. Using the elastic properties of the

pipe, the stiffness matrix for the piping system is determined. This includes the effects of torsional, bending, shear, and axial deformations, as well as the local flexibilities of piping curved members. Next, the frequencies and mode shapes for all the significant modes of vibrations are calculated. After the frequency is determined for each mode, the corresponding horizontal and vertical spectral accelerations with appropriate damping are read from the appropriate response spectrum curves. For each mode, the inertia response forces, moments, displacements and accelerations are determined due to excitation in the three directions simultaneously (two horizontal and one vertical). Finally, the stresses are determined by taking the SRSS of the individual components. The relative displacement effects between piping supports are discussed in Section 3.7.3.1.

3.7.3.8.2 Allowable Stresses

Allowable stresses in the piping caused by an earthquake are in accordance with Section III of the ASME Code. Allowable stresses in the earthquake restraint components, such as snubbers, are in accordance with any additional stress limits that may have been established by ASME Code, Section III at the time the restraint components were purchased.

3.7.3.9 Multiple Supported Equipment Components with Distinct Inputs

When the equipment or component is supported at points with different elevations, either the envelope of these elevation response spectra or multiple support excitation is used for the seismic qualification of the equipment.

3.7.3.10 Use of Constant Vertical Load Factors

In general, Seismic Category I subsystems are analyzed in the vertical direction using the methods specified in Section 3.7.3.1. No vertical static factors are used for subsystems.

3.7.3.11 Torsional Effects of Eccentric Masses

Piping systems are modeled to include projecting masses such as valve motor operators. The actual stiffness of the connecting member is not expected to influence the system appreciably. However, an approximation is made by assuming a member stiffness equal to that of the piping in which the valve is installed.

3.7.3.12   Piping Outside Containment Structure**3.7.3.12.1   Buried Piping**

Seismic design criteria for buried piping are as follows:

- A. Intake structure is designed such that the differential movement between this structure and the earth is negligible and the seismic response spectrum utilized is the ground surface response.
- B. Conformance to allowable structural and piping stresses after the line penetrates the Auxiliary Building is assured by the use of expansion joints.

An alternate design method is to use flexible seals as the lines pass through pipe sleeves in the structure.

Important factors considered are the flexibility, supports, and restraints of lines which are virtually anchored in earth but which penetrate a structure. A flexibility analysis of these lines is performed to demonstrate that the piping and structures are not overstressed under the additive differential movement of the earth and structure.

3.7.3.12.2   Above Ground Piping

Seismic design criteria and methods of accounting for the effects of differential movement of buildings on piping and penetrations are described in Sections 3.7.2.1.2 and 3.7.2.7.

3.7.3.13   Interaction of Other Piping with Category I Piping

The protection of Category I piping from possible adverse effects of other piping during an earthquake is accomplished by several methods. Specifically, these methods are:

- A. Category I lines are physically separated from other lines to the extent possible so that failure of a line has no effect on Category I lines.
- B. All Category I boundary valves are designed to meet seismic criteria. A valve always serves as a pressure boundary and constitutes the seismic to non-seismic boundary. If failure in the non-seismic portion of the system could cause loss of function of the safety system, then an appropriate automatic or remote manual operator would be used if the valve is open during normal reactor operation.

- C. The pressure boundary valve is protected by restraining or anchoring the non-seismic portion of the system as required.

3.7.3.14   Seismic Analysis of Reactor Internals, Core and CEDMs

3.7.3.14.1   Reactor Internals and Core

The seismic analyses of the reactor internals and core consist of a preliminary phase and a final phase. The preliminary phase is used to obtain preliminary design seismic loads and displacements in the vertical and horizontal directions. Separate analyses are performed in the horizontal and vertical directions. The final phase analyses are used to confirm that the design basis loads determined during the preliminary phase are conservative and result in an adequate design.

In the horizontal direction, because the relative displacements between the core and core shroud and between the core support barrel and pressure vessel snubbers are sufficiently large to close the gaps that exist between these components, a nonlinear horizontal time history analysis is performed. The horizontal nonlinear analysis is divided into two parts. In the first part, the internals and core are analyzed to obtain the internals' response and the proper dynamic input for the reactor core model. In the second part, the core plate motion from the first part is applied to a more detailed nonlinear model of the reactor core. The input excitation to the internals model is the response time-history of the reactor vessel at the internals support determined from the RCS analysis. Coupling effects between the internals and reactor vessel are accounted for by including a simplified representation of the internals with the RCS model. This is discussed in Section 3.7.2.

In the vertical direction, the preliminary analysis utilizes the linear modal analysis response spectrum method. The response spectra used are based upon the acceleration of the reactor vessel flange. If the preliminary linear vertical analysis indicates that the response of the core may be sufficiently large to cause it to lift off the core plate, a vertical nonlinear analysis of the internals is also performed.

In these analyses, two horizontal components and the vertical component of the seismic excitation are considered and the maximum responses for the three components are combined by the method of square root of the sum of the squares.

Closely spaced modes are considered in accordance with Regulatory Guide 1.92.

3.7.3.14.1.1 Mathematical Models

Equivalent multimass mathematical models are developed to represent the reactor internals and core. The mathematical models of the internals are constructed in terms of lumped masses and elastic-beam elements. At appropriate locations within the internals and core, points (nodes) are chosen to lump the weights of the structure. A sketch of the internals and core showing the relative node locations for the horizontal model is presented in Figure 3.7-35. The criterion for choosing the number and location of mass concentration is to provide for accurate representation of the dynamically significant modes of vibration of each of the internals components. Between the nodes properties are calculated for moments of inertia, cross-section areas, effective shear areas, and lengths. Separate horizontal and vertical models of the internals and core are formulated to more efficiently account for structural differences in these directions. In the horizontal nonlinear lumped mass representation of the internals and core, shown in Figure 3.7-36, gap and spring elements are used to represent contact between the fuel and core shroud. Lumped-mass nodes in the core are positioned to coincide with fuel-spacer grid locations. To simulate the nonlinear motion of the fuel, nonlinear spring couplings are used to connect corresponding nodes to the fuel assemblies and core shroud. Incorporated into these nonlinear springs is the spacer grid impact stiffness derived from test results. The core is modeled by subdividing it into fuel assembly groupings and choosing stiffness values to adequately characterize its beam response and contacting under dynamic loading.

The horizontal nonlinear reactor core model consisting of one row of 15 individual fuel assemblies is depicted in Figure 3.7-37. In this model each fuel assembly is represented with mass points located at spacer grid locations. To simulate the gaps in the core, nonlinear spring couplings are used to connect corresponding nodes on adjacent fuel assemblies and core shroud. The impact stiffness and impact damping (coefficient of restitution) parameters for the gap elements are derived from the impact tests which are described in Section 4.2. The spacer grid impact representation used for the analysis is capable of representing two types of fuel assembly impact situations. In the first type, only one side of the spacer grid is loaded. This type of impact occurs when the peripheral fuel assembly hits the core shroud, or when two fuel assemblies strike one another. The second type of impact loading occurs typically when the fuel assemblies pile up on one side of the core. In this case, the spacer grids are subjected to a through-grid compressive loading.

The fuel assemblies in the coupled core/internals model and the detailed core model are modeled with beam elements to represent the horizontal stiffness between mass points and rotational springs at each end to simulate the end fixity existing at the top and bottom of the core. The valve used for fuel horizontal stiffness and end fixity is based upon a parametric study in which analytic predictions are correlated with fuel assembly static and dynamic test data. Fuel assembly structural damping as a function of vibrational amplitude was derived from fuel assembly forced vibration and pluck tests defined in Section 4.2. The damping values used in the seismic analysis of the reactor internals are in accordance with the values in Table 3.7-1. Figure 3.7-38 shows the idealized linear vertical model. The vertical nonlinear model is shown in Figure 3.7-39.

Additional salient details of the internals and core models are discussed in the following paragraphs.

A. Hydrodynamic Effects

It has been shown both analytically and experimentally (Reference 2) that immersion of a body in a dense-fluid medium lowers its natural frequency and significantly alters its vibratory response as compared to that in air. The effect is more pronounced where the confining boundaries of the fluid are in close proximity to the vibrating body as in the case for the reactor internals. The method of accounting for the effects of a surrounding fluid on a vibrating system has been to ascribe the system additional or "hydrodynamic mass".

The hydrodynamic mass of an immersed system is a function of the dimensions of the real mass and the space between the real mass and confining boundary.

Hydrodynamic mass effects for moving cylinders in a water annulus are discussed in References 2 and 3. The results of these references are applied to the internals' structures to obtain the total (structural plus hydrodynamic) mass matrix that is then used in the evaluation of the natural frequencies and mode shapes.

B. Core Support Barrel

The core support barrel is modeled as a beam with shear deformation. It has been shown that the use of beam theory for cylindrical shells gives sufficiently accurate results when shear deformation is included (References 4 and 5).

C. Fuel Assemblies

The fuel assemblies are modeled as uniform beams with rotational springs at each end to represent the proper end condition. The member properties for the beam elements representing the fuel assemblies are derived from the results of experimental tests of fuel-assembly load deflection characteristics and fundamental natural frequency.

D. Support-Barrel Flanges

To obtain accurate lateral and vertical stiffness of the upper and lower core-support barrel flanges and the upper guide structure support barrel upper flange, finite-element analyses of these regions are performed. As shown in Figure 3.7-40, these areas are modeled with quadrilateral and triangular ring elements. Unit deflections and rotations are applied in the lateral and axial directions, and the resulting reaction forces are calculated. These results are then used to derive the equivalent member properties for the flanges.

E. Upper Guide Structure

For the horizontal model, the upper guide structure including CEA shrouds, connecting plates and tie rods are modeled as cantilever beams. A separate member is modeled to account for the connection between the tie rods and the upper guide structure support plate.

F. Lower Support Structure

To obtain vertical stiffness for the lower support structure grid beams and cylinder, a finite element analysis is performed. Displacements due to vertical (out-of-plane) loads applied at the beam junctions are calculated through the use of the computer code ANSYS, a description of which is given in Section 3.9.1.2.2.2. Average stiffness values based on these results yield an equivalent member cross-section area for the vertical model.

3.7.3.14.1.2 Analytical Techniques

A. Natural Frequencies and Mode Shapes

The mass- and beam-element properties of the models are utilized in the computer code ANSYS to obtain the natural frequencies and mode shapes. The program utilizes the

stiffness-matrix method of structural analysis. The natural frequencies and mode shapes are extracted from the system of equations.

$$(\underline{K} - W_n^2 \underline{M}) \phi_n = 0$$

where:

$\underline{K}$ = modal stiffness matrix

$\underline{M}$ = modal mass matrix

W_n = natural circular frequency for the n^{th} mode

ϕ_n = normal mode shape matrix for the n^{th} mode

The mass matrix, $\underline{M}$, includes the hydrodynamic and structural masses.

B. Response Calculation Methods

1. Response Spectra Method

The response spectrum analysis is performed using the modal extraction data and the following relationships for each mode:

a. Nodal Accelerations

$$\ddot{X}_{in} = \Gamma_n A_n \phi_{in} \quad (3.7-1)$$

where:

$\ddot{X}_{in}$ = absolute acceleration at node "i" for node "n"

Γ_n = modal participation factor

A_n = modal acceleration from response spectrum

ϕ_{in} = mode shape factor at node "i" for node "n"

b. Nodal Displacement

$$Y_{in} = \frac{\ddot{X}_{in}}{W_n^2} \quad (3.7-2)$$

where:

Y_{in} = displacement at node "i" for mode "n"
relative to base

W_n = natural circular frequency for n^{th} mode

2. Member Forces and Moments

$$F_n = \frac{\Gamma_n A_n}{W_n^2} \bar{F}_n \quad (3.7-3)$$

where:

F_n = actual member force for mode "n"

$\bar{F}_n$ = modal member force for mode "n"

The effect of the fluid environment is accounted for by defining the modal participation as follows:

$$\Gamma_n = \frac{\sum_{j=1}^M W_{si} \phi_{in}}{\sum_{i=1}^M \sum_{j=1}^M \phi_{in} W_{ij} \phi_{jn}} \quad (3.7-4)$$

where:

W_{si} = structural weight of node "i"

W_{ij} = structural + hydrodynamic weight terms

M = number of masses

The SRSS method is normally used to combine the modal responses. Where modal frequencies are closely spaced, the responses of these modes are combined by the sum of their absolute values. The modal damping factors are obtained by the method of "mass mode weighting", which gives:

$$B_n = \frac{\sum M_i \phi_{in} B_i}{\sum M_i \phi_{in}} \quad (3.7-5)$$

where:

B_n = modal damping factor

M_i = structural mass of mass node "i"

ϕ_{in} = absolute value of the mode shape at mass node "i"

B_i = damping associated with mass point "i"

C. Nonlinear Analysis

The nonlinear seismic response and impact forces for the internals and fuel are determined using the CESHOCK computer program (see Section 3.9.1.2.2.4). The computer program provides the numerical solution to transient dynamic problems by step-by-step integration of the differential equations of motion. The input excitation for the model is the time-history accelogram of the reactor vessel.

Input to the CESHOCK computer program consists of initial conditions, nodal lumped masses, linear-spring coefficients, mass moments of inertia, nonlinear spring curves, and the acceleration time-histories. The output from the CESHOCK computer program consists of displacements, translational and angular accelerations, impact forces, shears, and moments.

3.7.3.14.1.3 Analysis Procedures for Damping

The procedures used to account for damping in the analysis of the reactor internals and core are given in Section 3.7.3.14.1.2(B).2. Uniform modal damping factors are used in the analysis of other NSSS vendor supplied seismic subsystems.

3.7.3.14.1.4 Results

The nonlinear response loads for the internals, including impacting, if any exist, are determined for the vertical and horizontal directions. Loads for the fuel are determined in a separate reactor core nonlinear analysis. The results are determined for the safe shutdown earthquake (SSE) and the operational basis earthquake (OBE).

3.7.3.14.2 Control Element Drive Mechanisms (CEDM)

The pressure-retaining components of the CEDM are designed to the appropriate stress criteria of ASME Code Section III for all

loadings specified. The structural integrity of the CEDM when subjected to seismic loadings is verified by combination of test and analysis. Methods of modal dynamic analysis employing response spectrum techniques or time history analysis are supported with experimentally obtained information.

3.7.3.14.2.1   Input Excitation Data

For the dynamic analyses, a response spectra or time history definition of the excitation at the base of the CEDM nozzle is obtained from the seismic analysis of the RCS. The excitation is applied simultaneously in three mutually perpendicular directions (two horizontal and one vertical).

3.7.3.14.2.2   Analysis

A dynamic analysis of the mathematical structural model is performed utilizing one or more of the computer programs discussed in Section 3.9.1.2.

3.7.3.14.2.3   Functional Test

A functional test utilizing a minimum drop weight is performed to verify that drop characteristics meet the input design requirements. Results from this test are compared to the calculated CEDM deflections under seismic loading for the individual site. Verification of the proper function is thus established based on both analytical and test results.

3.7.3.15   Analysis Procedures for Damping

3.7.3.15.1   Subsystems Other Than NSSS

The analysis procedure used to account for the damping in non-NSSS Subsystems complies with Section 3.7.2.15.

3.7.3.15.2   Nuclear Steam Supply System

The procedures used to account for damping in the analysis of the reactor internals and core are given in Section 3.7.3.14. Uniform modal damping factors are used in the analysis of other NSSS vendor supplied seismic subsystems.

The analytical method for evaluating the faulted condition uses a linear elastic model as described in Section 3.7.3. The ASME Section III allowable stress limits are met for faulted loads, including the safe shutdown earthquake and system transient loads described in Section 3.7.1.

3.7.4 SEISMIC INSTRUMENTATION

3.7.4.1 Comparison with Regulatory Guide 1.12

Seismic instrumentation is necessary to determine promptly the seismic response of nuclear power plant features to permit comparison of such response with that used as the design basis.

The seismic instrumentation utilizes multiple types of sensor-recorders with a playback capability available in the control room area. The location and function of these seismic devices are selected to provide adequately for the determination of seismic event loads into the structures via computerized analysis programs. The seismic instrumentation program is in accordance with Regulatory Guide 1.12.

3.7.4.2 Location and Description of Instrumentation

3.7.4.2.1 Active Instruments

A central recording time-history accelerograph unit with strip chart recorder, indicator lights, and playback system are provided in the control room area. The lights indicate whether the system is triggered and whether the operating basis or safe shutdown maximum accelerations and response spectra are exceeded in any one of the three orthogonal directions on the basemat of the containment structure. These orthogonal directions will coincide with the major axes of the analytical model to be used in the seismic analysis of the plant structure. I

The central unit connects to:

- A. A triaxial trigger placed on the reactor building foundation slab. At an acceleration of 0.02g in any direction, the central unit and attached accelerographs is activated. This acceleration level is chosen to screen out minor disturbances.
- B. Triaxial time-history accelerographs, each of which measures the absolute acceleration as a function of time in three orthogonal directions: these directions coincide with the major axes of the analytical model of the structure. These accelerographs are placed at the following locations:
  - 1. on the common basemat,
  - 2. on the shield building wall, and
  - 3. a free field

Specific instrument locations are chosen to account for effects of specific site conditions on structural response (e.g., rocking of structure).

C. Response Spectrum Recorders

A triaxial response spectrum recorder capable of measuring both the horizontal and vertical motions is provided at each of the following locations:

1. on the common basemat, adjacent to the time-history accelerograph. Control room indication is provided for this location. This location serves the dual purpose of monitoring the base slab response and the support motion of reactor equipment;
2. on a floor in the Nuclear Annex; and
3. on a selected Seismic Category I piping support.

These locations are chosen to allow meaningful correlation between the recorded accelerations and those calculated using the analytical model of the structure.

3.7.4.2.2 Passive Instruments

A triaxial peak recorder which measures the absolute peak acceleration in three orthogonal directions coinciding with the major axes of the analytical model are provided at each of the following locations:

- A. on the safety injection tank in the steel containment vessel;
- B. on the safety injection piping in the steel containment vessel; and
- C. on the essential service return piping in the Nuclear Annex.

3.7.4.3 Control Room Operator Notification

The centrally located seismic indicating and recording equipment in the main control room is the source of operator information concerning the acknowledgment of an earthquake. An acceleration of 0.02g in any direction activates the seismic trigger which turns on the seismic monitors and lights up the seismic alarm lights at the central station panel.

An operator can call up triaxial response spectral recordings from the common basemat slab monitor. The operator also can observe the time-history accelerograph via its playback capability to evaluate the indications from the common basemat monitor.

Observed values which exceed OBE acceleration threshold on the time-history accelerograph are indicated by an alarm light during playback. Further analysis is needed to authenticate structural loads and to evaluate observations via the structural response-seismic model. An observation that exceeds the SSE acceleration threshold is validated in a similar manner with the structural response-seismic model. When evaluated accelerations exceed OBE threshold values, the reactor is manually shutdown. The alarm lights and the recorder data are available simultaneously with the seismic event.

3.7.4.4   Comparison of Measured and Predicted Responses

The computer program which evaluates the time-history data computes the maximum response accelerations at various points of the model. The observed response spectra are compared with the computed response spectra. Agreement between the observed response spectra and the computed response spectra from the time-history inputs demonstrates the adequacy of the analytical model. The magnitude of actual forces at various structural locations is then compared to design values to authenticate the capability of the plant to continue operation without undue risk to the health and safety of the public.

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TABLE 3.7-1**DAMPING VALUES****

Structure	Operating Basis Earthquake (Percent of Critical)	Safe Shutdown Earthquake (Percent of Critical)
Welded steel structures	2.0	4.0
Bolted steel structures	4.0	7.0
Prestressed concrete structures	2.0	5.0
Reinforced concrete structures	4.0	7.0
Equipment (steel assembly)	2.0	3.0
Piping* (diameter <12 inches)	1.0	2.0
Piping* (diameter >12 inches)	2.0	3.0

* When response spectra method of analysis is used, damping values may be based on Code Case N-411-1.

** Soil material dampings are provided in Section 2.5.2.

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE A-1

Figure
3.7-1

Amendment I
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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE A-1

Figure
3.7-2

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE B-1

Figure

3.7-3

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE B-1

Figure
3.7-4

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE B-2

Figure
3.7-5

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE B-2

Figure
3.7-6

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE B-3

Figure
3.7-7

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE B-3

Figure
3.7-8

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE B-4

Figure
3.7-9

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE B-4

Figure
3.7-10

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE C-1

Figure

3.7-11

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December 21, 1990

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE C-1

Figure
3.7-12

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE C-2

Figure
3.7-13

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE C-2

Figure

3.7-14

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December 21, 1990

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE C-3

Figure
3.7-15

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE C-3

Figure
3.7-16

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE D-1

Figure
3.7-17

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE D-1

Figure

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE B-1.5

Figure
3.7-19

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE B-1.5

Figure
3.7-20

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE B-3.5

Figure
3.7-21

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE B-3.5

Figure

3.7-22

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT GROUND SURFACE FOR CASE C-1.5

Figure

3.7-23

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CALCULATED HORIZONTAL & VERTICAL SPECTRA
AT FOUNDATION LEVEL FOR CASE C-1.5

Figure

3.7-24

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SELECTED SMOOTH SPECTRUM AND
SPECTRUM FOR SYNTHETIC TIME HISTORY H2

Figure

3.7-26

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SELECTED SMOOTH SPECTRUM AND
SPECTRUM FOR VERTICAL SYNTHETIC TIME HISTORY V

Figure
3.7-27

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STICK MODEL OF INTERNAL STRUCTURE
(FOR HORIZONTAL ANALYSIS)

Figure
3.7-28

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STICK MODEL OF SHIELD BUILDING

Figure

3.7-29

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SYSTEM 80 ^{TM} +

FINITE ELEMENT MODEL OF STEEL CONTAINMENT VESSEL
(FOR SSI ANALYSIS)

Figure
3.7-30

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SCHEMATIC REPRESENTATION OF COMBINED
STRUCTURAL MODEL OF RB
(SSI ANALYSES; FLOOR ECCENTRICITIES NOT SHOWN)

Figure
3.7-31

COMPONENT NAME	MASS POINT NUMBER	DEGREES OF FREEDOM	COMPONENT NAME	SUPPORT POINT NUMBER	RESTRAINT
REACTOR	9996, 9912, 9908 9995	X, Y, Z X, Y	REACTOR	1999 2999 4999, 5999 1910, 2910, 4910, 5910	FX, FY FIXED
STEAM GENERATORS	404, 3404, 412, 3412 409, 3409 408, 3408	X, Y, Z X, Y X, Y	STEAM GENERATORS	70, 3070 250, 3250 211, 221 3211, 3221	FX, FZ FY FX FX
PUMPS	1103, 2103, 4103, 5103 1101, 2101, 4101, 5101	X, Y, Z X, Y, Z	REACTOR COOLANT PUMPS (TYPICAL)	11, 15, 21, 25 51, 55, 61, 65 71	FZ FX, FY FX, FY
REACTOR COOLANT PIPING	800, 3800 1760, 2760, 4760, 5760 1580, 2580, 4580, 5580	X, Z X, Y, Z X, Y, Z			

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REACTOR COOLANT SYSTEM SEISMIC ANALYSIS MODEL

Figure

3.7-32

- MASS POINT
- STRUCTURAL JOINT
- }- SUPPORT POINT

MASS POINT	DEGREE OF FREEDOM
6110, 6115, 6120, 6125, 6130	X, Y
6135	X, Y, Z

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PRESSURIZER SIESMIC ANALYSIS MODEL

Figure
3.7-33

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	SURGE LINE SEISMIC ANALYSIS MODEL	Figure 3.7-34

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REACTOR INTERNALS HORIZONTAL SEISMIC ANALYSIS MODEL

Figure

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REACTOR INTERNALS
NONLINEAR HORIZONTAL SEISMIC MODEL

Figure

3.7-36

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CORE SEISMIC MODEL ONE ROW OF 15 FUEL ASSEMBLIES

Figure

3.7-37

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REACTOR INTERNALS LINEAR VERTICAL SEISMIC MODEL

Figure
3.7-38

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REACTOR INTERNALS NONLINEAR VERTICAL SEISMIC MODEL

Figure

3.7-39

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CORE-SUPPORT BARREL
UPPER FLANGE FINITE-ELEMENT MODEL

Figure
3.7-40

GENERAL NOTE: APPLICABLE TO BOTH OBE AND SSE, INDEPENDENT OF PIPE DIAMETER.

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DAMPING VALUE FOR SEISMIC ANALYSIS OF PIPING

Figure

3.7-41

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APPENDIX 3.7A

COUPLED REACTOR COOLANT SYSTEM SEISMIC RESULTS

I

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3.7A-2	Load Tables for Reactor Coolant System	

APPENDIX 3.7A

COUPLED REACTOR COOLANT SYSTEM SEISMIC RESULTS

OVERVIEW

This Appendix contains the results of the coupled Reactor Coolant System (RCS) seismic analyses.

Table 3.7A-1 contains the RCS coupled frequencies and modes (refer to Figures 3.7-32 and 3.7-33 for directions and joint locations).

Table 3.7A-2 contains the seismic loads due to the SSE.

I

TABLE 3.7A-1

**NATURAL FREQUENCIES AND DOMINANT
DEGREES OF FREEDOM FOR SIGNIFICANT
MODES OF THE COUPLED REACTOR COOLANT SYSTEM**

Mode	Freq. (Hertz)	Dominant Degrees of Freedom		
		Joint Number	Direction	Location
1	1.7	9912	X	RV INT
2	1.7	9912	X	RV INT
7	8.8	404, 3404	Y	SG
8	9.6	9996	X	RV
10	10.9	9996	X	RV
11	11.2	9996	Y	RV
15	12.2	404, 3404	Y	SG
18	12.5	1103, 2103, 4103, 5103	X, Y	RCP Motor
20	13.3	1103, 2103, 4103, 5103	X, Y	RCP Motor
25	14.5	1103, 2103, 4103, 5103	Y	RCP Motor
27	15.4	9996, 409, 3409	X	RV, SG
28	15.5	408, 3408	X	SG INT
32	17.6	1103, 1101, 2103, 2101, etc.	Z	RCP
37	18.1	9912, 9996	Z	RV INT, RV
40	19.0	404, 3404	Z	SG
42	19.1	9912, 9996, 412, 3412	Z	RV INT, RV, SG INT
44	19.8	9995	Y	RV
45	20.0	1101, 2101, 4101, 5101	X	RCP
49	20.7	1101, 2101, 4101, 5101	X	RCP
50	21.0	9995	Y	RV
51	21.2	404	Z	SG
52	21.4	3404	Z	SG
53	21.9	1101, 2101, 4101, 5101	X	RCP
56	24.7	404, 3404	X	SG
57	25.2	404, 3404	X	SG
60	26.6	6120	X	PRESS
62	26.7	6120	Y	PRESS
64	29.0	6120	X	PRESS
65	30.3	6120	Y	PRESS
100	42.8	6120	Z	PRESS

TABLE 3.7A-2

(Sheet 1 of 4)

LOAD TABLES FOR REACTOR COOLANT SYSTEM

Seismic Excitation - SSE		Seismic Loads, Kips and Ft-Kips	
<u>Support Location</u>	<u>Reaction Component</u>	<u>Calculated Maximum</u>	<u>Design Specification</u>
Steam generator upper key	Fx	601	1800
Steam generator snubber assembly	Fy	913	1500
Steam generator vertical pad	Fz (1,3)	359	560
	Fz (2,4)	1287	2060
Steam generator holddown bolt	Fz (1,3)	-	550;605
	Fz (2,4)	-	675
Steam generator lower key	Fx	422	1100
Reactor vessel horiz. column support	Fc	1361	3000
Reactor vessel column base	Fa	13	25
	Fb	1138	2000
	Fc	360	750
	Ma	355	800
	Mb	567	1300
	Mc	111	200
Pump vertical column	Fz	132	275
Pump snubber	Fa	360	650
Pump upper horizontal column	Fa	178	300
Pump lower horizontal column	Fa	88	190
Pressurizer key	Fk	90	110
Pressurizer support skirt	Fv	240	280
	Fh	236	280
	Mt	5	50
	Mb	1592	1900

TABLE 3.7A-2 (Cont'd)**(Sheet 2 of 4)****LOAD TABLES FOR REACTOR COOLANT SYSTEM**

Seismic Excitation - SSE		Seismic Loads, Kips and Ft-Kips	
<u>Support Location</u>	<u>Reaction Component</u>	<u>Calculated Maximum</u>	<u>Design Specification</u>
Reactor vessel inlet nozzle	Fa	98	720
	Fb	41	120
	Fc	96	650
	Ma	286	700
	Mb	301	800
	Mc	255	500
Reactor vessel outlet nozzle	Fa	662	1300
	Fb	239	400
	Fc	87	165
	Ma	210	650
	Mb	611	1250
	Mc	1788	2500
Reactor vessel column upper flange	Fa	13	25
	Fb	1138	2000
	Fc	23	110
	Ma	345	1400
	Mb	31	140
	Mc	127	200
Reactor vessel lower key	Fc	355	520
Steam generator inlet	Fa	665	1700
	Fb	219	950
	Fc	95	950
	Ma	444	1100
	Mb	434	1700
	Mc	918	1700
Steam generator support skirt	Fx	801	1860
	Fz	2699	3300
	Mx	1957	2400
	My	2562	4800
	Mz	419	900

TABLE 3.7A-2 (Cont'd)**(Sheet 3 of 4)****LOAD TABLES FOR REACTOR COOLANT SYSTEM**

Seismic Excitation - SSE		Seismic Loads, Kips and Ft-Kips	
<u>Support Location</u>	<u>Reaction Component</u>	<u>Calculated Maximum</u>	<u>Design Specification</u>
Steam generator outlet nozzle	Fa	27	100
	Fb	43	190
	Fc	58	190
	Ma	222	370
	Mb	244	1000
	Mc	209	1000
Pump inlet nozzle	Fx	54	100
	Fy	47	110
	Fz	27	60
	Mx	244	520
	My	225	420
	Mz	141	550
Pump outlet nozzle	Fa	126	550
	Fb	35	110
	Fc	17	80
	Ma	112	300
	Mb	160	350
	Mc	412	1280
Pump skirt/casing interface	Fx	147	280
	Fy	89	190
	Fz	483	540
	Mx	239	750
	My	151	500
	Mz	40	170
Pump motor support upper flange (latern top)	Fx	106	530
	Fy	127	600
	Fz	198	240
	Mx	1579	4250
	My	1915	3750
	Mz	123	200

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TABLE 3.7A-2 (Cont'd)**(Sheet 4 of 4)****LOAD TABLES FOR REACTOR COOLANT SYSTEM**

Seismic Excitation - SSE		Seismic Loads, Kips and Ft-Kips	
<u>Support Location</u>	<u>Reaction Component</u>	<u>Calculated Maximum</u>	<u>Design Specification</u>
Pump motor support lower flange (latern bottom)	Fx	106	280
	Fy	127	280
	Fz	198	240
	Mx	1273	4100
	My	1171	2625
	Mz	123	210
Piping at reactor vessel inlet nozzle	M max	488	2000
Piping at reactor vessel outlet nozzle	M max	1902	4834
Piping at steam generator inlet nozzle	M max	1109	4834
Piping at steam generator outlet nozzle	M max	391	2000
Piping at pump inlet nozzle	M max	361	2000
Piping at pump outlet nozzle	M max	456	2000

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APPENDIX 3.7B

**SOIL STRUCTURE INTERACTION (SSI) ANALYSIS
METHODOLOGY AND RESULTS**

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APPENDIX 3.7B**SOIL STRUCTURE INTERACTION (SSI) ANALYSIS
METHODOLOGY AND RESULTS****OVERVIEW**

This appendix describes the SSI methodology and presents analysis results used to establish seismic design loads for the Reactor Building (RB) and Reactor Coolant System (RCS) of the System 80+ Standard Design. The analyses were performed based on a Safe Shutdown Earthquake (SSE) excitation of 0.30g and an Operating Basis Earthquake (OBE) of 0.10g horizontal peak ground accelerations at rock outcrop. A set of nine soil profiles developed in Reference 7 of Section 3.7 to represent generic site conditions were used as the soil medium in the SSI analysis. The SSI analysis results are provided in the form of in-structure response spectra corresponding to major elevations, and internal resisting forces at each floor of the RB.

A fixed-base analysis case with no embedment considerations and no SSI effects was also performed using the rock outcrop motion as direct input excitation to the RB foundation. To expedite the generation of OBE design spectra, two SSI analyses and a fixed-base analysis were performed and, based on the ratio of the structural response between the three OBE analyses and the corresponding SSE analyses, generic scaling factors were derived to scale all SSE in-structure response spectra and develop OBE spectra. The OBE SSI cases were selected based on the critical SSE results.

The SSI analyses are performed using a Power Generation Complex (PGC) with dual foundation. After these analyses were completed, the foundation design was modified to a common basemat that founds all PGC structures (RB and adjacent structures). To evaluate the impact of the foundation modification on the SSI results, one critical SSI case is reanalyzed with a "common basemat foundation". The results of the "common basemat" analysis are compared to the original "dual foundation" results in order to determine whether the latter envelop the "common basemat" results and can be used in final design.

1.0 ANALYSIS METHODOLOGY

A standard plant design must be based on a sequence of analyses that cover a broad range of sites that the plant might be built on. Each site has unique seismic response characteristics. Therefore, the investigation of multiple sites accounts for the likelihood of resonance between the RB and the site soil. The sites selected for the RB SSI analyses have free-field amplifications that cover a broad range of frequencies with which fundamental structural frequencies may coincide. Hence, the envelope of the results provide the maximum seismic response to the SSE and OBE rock outcrop motions when the RB is founded on soil sites that are bounded by the selected soil profiles.

1.1 SASSI FORMULATION

For the RB SSI analyses, the methodology of the computer program SASSI was used. SASSI (System for Analysis of Soil-Structure Interaction, Reference 6 of Section 3.7) is the most versatile tool currently available for SSI industry practice. The SASSI program uses a general substructuring method, which is formulated in the frequency domain using the complex response method and the finite element technique. In a substructuring method, the soil strata and halfspace are analyzed first, in the frequency domain, and the impedance and scattering properties at the soil-structure interface are established. Subsequently, these properties are used as boundary conditions in a dynamic analysis of the structure with a loading that depends on the free-field motions.

According to the SASSI formulation, the solution of the SSI problem reduces to three steps:

- o Solution of the site response problem to determine the free-field motions within the embedded part of the structure.
- o Solution of the impedance and scattering problem.
- o Solution of the structural problem. This involves forming the complex stiffness matrices and load vector and solving the equations of motion for the final displacements.

SASSI is structured in a modular form. The code is segmented in independent subprograms (modules) which are executed sequentially. Each of these modules performs one of the tasks required in the sequence of the analysis of SSI. The modules are:

- o SITE
- o POINT
- o HOUSE

- o AXSYM
- o MOTOR
- o ANALYS
- o COMBIN
- o MOTION
- o STRESS

It is not necessary to execute all modules for a given SSI analysis. The modules that are executed depend on the type of problem (seismic analysis or forced vibration analysis), the type of model (2-D, 3-D or axisymmetric) and the kind of results desired (transfer functions, accelerations, velocities, displacements).

A schematic of the SASSI approach used in this analysis is shown in Figure 3.7B-1. Each step of this approach is described below.

1.2   SITE RESPONSE

The site model consists of horizontal soil layers overlying a rigid base (Figure 3.7B-2). All material properties are visco-elastic. Soil unit weights were obtained from Section 2.5.2. The stiffness properties of each layer consist of shear modulus and Poisson's ratio. The damping properties of each layer consist of material damping ratios associated with shear waves (S-waves) and compression waves (P-waves). Stiffness and damping are compatible with the strains induced in the soil by the earthquake excitation. The stiffness and damping properties for the soil material were obtained from the strain iterated properties contained in Tables 2C-1 through 2C-16 of Appendix 2C. The site response and the seismic input are generated using the module SITE.

In the horizontal analyses, the seismic excitation is in the form of vertically propagating S-waves. In the vertical analyses, the seismic excitation is in the form of vertically propagating P-waves. The control motion is specified at the free-field ground surface and includes the site amplification effects for each particular site. The results of the solution of the site response problem were used to obtain the frequency-dependent impedance and scattering functions.

1.3   IMPEDANCE AND SCATTERING COMPUTATION

Because of the size and shape of the RB foundation, an axisymmetric approach was used to obtain the foundation impedances and scattering matrices. This was accomplished with the module AXSYM.

The stiffness and mass matrices for the axisymmetric soil-foundation system are first produced by HOUSE using an axisymmetric 4-node solid finite element mesh, as shown in Figure 3.7B-3. The model in Figure 3.7B-3 consists of axisymmetric solid elements with appropriate material properties connected to semi-infinite layered zones which are represented by axisymmetric transmitting boundaries. Depth of soil to bedrock varies according to the case analyzed. Figure 3.7B-4 shows a three-dimensional representation of the transmitting boundaries and the semi-infinite soil layers. There were two foundations modeled: the foundation of the RB (220 ft. diameter) and a surrounding foundation (with 111 ft. inner radius and 210 ft. outer radius) representing the foundation of all the adjacent buildings. The two foundations are not connected and they can move independent of each other. The foundation of the adjacent structures is connected to the side soil with a rigid wall, while the RB foundation is free all around.

The stiffness and mass matrix information along with the information from SITE were then provided as input to the module AXSYM. This module generated the impedance matrix and the scattering properties corresponding to P-wave in vertical direction, S-wave in one horizontal direction, and S-wave in the orthogonal horizontal direction. Since the foundations are rigid, AXSYM computed impedances and scattering properties corresponding to 6 degrees-of-freedom (DOFs) for each foundation: two horizontal translations, one vertical translation, two rocking rotations, and one torsional rotation. The final impedance matrix consisted of a 12 x 12 complex matrix to account for foundation-to-foundation interaction. The scattering matrix was a 12 x 3 complex matrix accounting for the variation in motion due to the imbedment, size and shape of the foundation, and the properties of the soil layers. The dimensions of the matrix are based on the 12 DOFs of the two foundations and the three directions of input motion.

1.4 STRUCTURAL ANALYSIS

Following the impedance computation, the stiffness and mass information of the general 3-D superstructures (RB and adjacent buildings) were generated using the module HOUSE again. The solution of the combined system (soil, foundation and superstructures) was carried out in the module ANALYS. The latter generated frequency-dependent transfer functions for every node in the superstructure models.

The transfer functions were subsequently multiplied by the Fourier Transform of the control motion (in this case, the free-field surface motion) to obtain the Fourier Transforms of the response acceleration time histories of all nodal points.

Using an Inverse Fourier Transform technique, response acceleration time histories were obtained in the time domain. This operation was performed by the module MOTION.

The time histories of the input control motions and the output (response) consisted of 4096 acceleration values at a time step of 0.005 sec. The input control motions were obtained from Reference 7 of Section 3.7.

2.0   SSI ANALYSIS CASES

The SSI analyses were performed for both SSE and OBE. A summary of all SSI analysis cases is presented in Table 3.7B-1. For the SSE, nine SSI analyses were performed using all generic soil profiles developed in Reference 7 of Section 3.7. For the OBE, two SSI analyses were performed using the most critical generic soil profiles (B3.5 and B4).

All analyses are three-dimensional with input excitation provided in three directions simultaneously. The generic soil sites differ from each other with respect to soil properties and depth of soil over bedrock. There are five cases with depth of soil to bedrock of 100 feet and four cases with soil depth to bedrock of 200 feet. The embedment depth of the RB is the same (52 feet) in all cases.

Because the analyses were performed in the frequency domain, the transfer functions were generated up to a maximum "cutoff" frequency. Cutoff frequencies are the maximum frequencies that the soil media can transmit without loss of accuracy in the solution. In the analyses, cutoff frequencies were computed based on the dimensions of the soil discretization. The maximum frequency that a soil layer can transmit corresponds to a wavelength equal to $h/5$, where h is the layer thickness. If the maximum frequency is found to be higher than 40 Hz, a cutoff frequency of 40 Hz is selected. This is consistent with the fact that the rock outcrop input motion has no frequency content beyond 40 Hz. Cutoff frequencies vary according to the soil profiles used in the analyses. Table 3.7B-12 summarizes the cutoff frequencies for all analysis cases.

The soil layers used in the SSI models and their associated properties are shown in Tables 3.7B-2 through 3.7B-10 for all cases respectively.

2.1   ANALYSIS RESULTS

The response acceleration time histories from the fixed-base and SSI analyses were used to compute in-structure response spectra for two damping values: 2 and 5%. Response spectra were

computed at 141 frequencies, the selection of which more than adequately satisfies the recommendations of the Standard Review Plan Section 3.7.1. The 141 frequencies include the recommended frequencies in the Standard Review Plan, additional frequencies corresponding to the main natural frequencies of the structural models, and frequencies beyond 34 Hz to account for the high frequency content of the motions. Table 3.7B-11 tabulates the frequency values at which response spectra were computed. For design purposes, response spectra broadened by $\pm 15\%$ and smoothened, according to Reg. Guide 1.122.

In-structure response spectra were generated at locations shown in the table below. Refer to Reference 8 of Section 3.7 for plots of the SSE and OBE spectra at all locations shown below.

Structure	Elev. (ft.)	Node	Location	Directions
				NO. (X=0-180, Y=90-270, Z=Vert)
RB FDTN	50.00	131	Center of foundation	X, Y, Z
SCV*	174.37	25	Midheight, 0-180 dir.	X, Y, Z
SCV	174.37	34	Midheight, 90-270 dir.	X, Y, Z
SCV	257.00	61	Top of SCV shell	X, Y, Z
SB**	261.88	125	Top of SB shell	X, Y, Z
IS***	64.73	141	Second Floor (C.M.)	X, Y, Z
IS	90.25	150	Third Floor (C.M.)	X, Y, Z
IS	91.82	155	SCV Support (C.M.)	X, Y, Z
IS	114.06	169	Fourth Floor (C.M.)	X, Y, Z
IS	142.92	183	Operating Floor (C.M.)	X, Y, Z
IS	207.48	210	Top of Crane Wall (C.M.)	X, Y, Z

* Steel Containment Vessel

** Shield Building

*** Internal Structure

**** Center of Mass

All listed nodes of the IS correspond to the centers of mass at the respective elevations.

2.2 SAFE SHUTDOWN EARTHQUAKE ANALYSIS RESULTS

Representative plots of the SSE response spectra are presented in Figures 3.7B-5 through 3.7B-12. The corresponding locations are at the RB foundation (Node 131), the top of the IS (Node 210), the top of the SCV (Node 61) and the top of the SB (Node 125). The spectra presented are raw spectra; i.e., without broadening and smoothening. The damping ratio is 2% of critical.

These plots are a comparison between results at the same location for all soil cases and the fixed-base. The following superimposed spectra are plotted:

<u>Figure No.</u>	<u>Building</u>	<u>Location</u>	<u>Direction</u>
3.7B-5	RB	Ftdn (Node 131)	X (0-180)
3.7B-6	RB	Ftdn (Node 131)	Z (vertical)
3.7B-7	IS	Top (Node 210)	X (0-180)
3.7B-8	IS	Top (Node 210)	Z (vertical)
3.7B-9	SB	Top (Node 125)	X (0-180)
3.7B-10	SB	Top (Node 125)	Z (vertical)
3.7B-11	SCV	Top (Node 61)	X (0-180)
3.7B-12	SCV	Top (Node 61)	Z (vertical)

At the foundation, the horizontal and vertical spectra show a wide scatter of the maximum peaks which are mostly accounted for by the variations in the free-field motion at the foundation level. The same phenomenon is observed at the top of the IS for the vertical spectra. However, in the IS horizontal spectra, most of the peaks are concentrated in the 3-6 Hz and the 9-20 Hz ranges reflecting amplification due to rocking of the structure (at the low frequency range) and amplification near the main horizontal natural frequencies of the IS (at the high frequency range). At the top of the SB, there is also a scatter in the results for both horizontal and vertical spectra, with maximum amplitudes occurring near the 5 Hz range for the horizontal, and near 10-15 Hz for the vertical. At the top of the SCV, most horizontal spectra have maximum amplitudes in the vicinity of 5 Hz, while the vertical spectra have maximum amplitudes in the 10-20 Hz range.

The range of site parameters used in the SSI analyses cover a broad range of site conditions. Soil amplification occurs at frequencies in the range of dominant structural modal frequencies. Therefore, resonance effects between the soil and the structures are captured in the SSI analyses and they are reflected in the results. As such, the combined SSI results ensure that adequate seismic loads for the System 80+ Power Generation Complex (PGC) have been generated for sites that are compatible with the generic sites used in these analyses.

2.3 OPERATING BASIS EARTHQUAKE ANALYSIS RESULTS

Spectral amplifications occur at the same frequencies as the equivalent SSE spectra. This is expected since the OBE soil profiles are identical to the SSE profiles of cases B3.5 and B4 respectively, with the exception of damping. Since soil damping and structural damping is lower for the OBE, the amplitudes of the spectral spikes are bigger than 1/3 of the respective SSE amplitudes. To evaluate a generic scaling factor to scale SSE results, spectral ratios of OBE vs. SSE were plotted as a function of frequency, for selected locations, as follows:

<u>Figure No.</u>	<u>Building</u>	<u>Location</u>	<u>Direction</u>
3.7B-13	RB	Ftdn (Node 131)	X (0-180)
3.7B-14	RB	Ftdn (Node 131)	Y (90-270)
3.7B-15	RB	Ftdn (Node 131)	Z (vertical)
3.7B-16	IS	Top (Node 210)	X (0-180)
3.7B-17	IS	Top (Node 210)	Y (90-270)
3.7B-18	IS	Top (Node 210)	Z (vertical)
3.7B-19	SB	Top (Node 125)	X (0-180)
3.7B-20	SB	Top (Node 125)	Y (90-270)
3.7B-21	SB	Top (Node 125)	Z (vertical)
3.7B-22	SCV	Top (Node 61)	X (0-180)
3.7B-23	SCV	Top (Node 61)	Y (90-270)
3.7B-24	SCV	Top (Node 61)	Z (vertical)

In general, the spectral factors fall in the range between 0.33-0.50. Factors corresponding to the frequency range between 0.1 Hz and (approximately) 3 Hz fall below 0.40. Factors that are between 0.40-0.50 correspond to the high frequency range between 3-100 Hz. Except for the SCV, there are few peaks exceeding 0.50. At the SCV, the factors exceed 0.50 mostly for the fixed-base case, while the SSE cases show only a few exceedances above 0.50.

The above observations led to the selection of generic factors that were different for each structure in the PGC, as shown below. At each structure, the generic factor is nearly an upper bound of all spectral ratios, with only a few isolated exceedances over the entire frequency range. Because the generic factors are selected from the critical SSE cases and they

constitute an upper bound for the majority of the frequencies, the application of these factors results in adequate and slightly conservative OBE seismic loads for design purposes.

<u>Structure</u>	<u>Direction</u>	<u>Factors</u>
PGC Foundation	X, Y, Z	0.4 (all frequencies)
IS (all elevations)	X, Y, Z	0.45 (all frequencies)
SB (all elevations)	X, Y, Z	0.45 (all frequencies)
SCV (all elevations)	X, Y	0.40 for frequencies $\leq$ 5 Hz 0.45 for frequencies $>$ 5 Hz
SCV (all elevations)	Z	0.40 for frequencies $\leq$ 10 Hz 0.65 for frequencies $>$ 10 Hz

3.0 SSI ANALYSIS COMMON BASEMAT CASE

To evaluate the impact of the modification from a dual foundation to a common basemat for all PGC structures, one critical SSI case is reanalyzed with a common basemat foundation.

Case B3.5 is selected for the "common basemat" analysis because, when the RB is coupled with the B3.5 soil profile, it is subjected to high accelerations which result in critical spectral peaks. This response is observed in the "dual foundation" results.

To reanalyze the B3.5 case, the PGC model is modified as shown in Figure 3.7B-25. The foundation is modeled as a continuous rigid basemat with rigid sidewalls which are in direct contact with the side soil. The adjacent-to-the-RB structures are connected to the center of the common basemat with rigid links.

The results of the "common basemat" analysis are documented in detail in Reference 25 of Section 3.7. A comparison of response spectra at selected locations is shown in Figures 3.7B-26 to 3.7B-33, as follows:

<u>Figure</u>	<u>Building</u>	<u>Location</u>	<u>Direction</u>
3.7B-26	RB	Fdtn (Node 131)	X (0-180)
3.7B-27	RB	Fdtn (Node 131)	Z (vertical)
3.7B-28	IS	Top (Node 210)	X (0-180)
3.7B-29	IS	Top (Node 210)	Z (vertical)
3.7B-30	SB	Top (Node 125)	X (0-180)
3.7B-31	SB	Top (Node 125)	Z (vertical)
3.7B-32	SCV	Top (Node 61)	X (0-180)
3.7B-33	SCV	Top (Node 61)	Z (vertical)

Because of the common basemat, the RB has direct interaction with the side soil. This means that the restraint that the side soil provides to the RB's seismic motions is no longer a secondary effect (as in the case of the dual foundation system), but a direct effect on the response of the RB. This causes "stiffening" of the system, which is clearly observed in the shift of the fundamental frequency of the response at all locations. As a direct consequence of the frequency shift in the horizontal directions of excitation, the horizontal response spectra show that the spectral peaks occur at higher frequency than that observed in the "dual foundation" spectra. Spectral amplitudes have, in general, decreased (compared to the dual foundation). The decrease in amplitude is caused by the additional restraint provided by the side soil, in both the translational and rocking modes. It must also be noted that in the IS, the vertical spectra corresponding to the common basemat design are almost enveloped by the spectra corresponding to the dual foundation. All ZPA (at all locations in three directions) are lower in the "common basemat" spectra. This is again a result of the additional restraint provided by the side soil to the PGC basemat. Internal forces in the IS and SB are also lower for the common basemat design (Reference 9 of Section 3.7).

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The analysis results have shown that the response of the RB with common basemat is enveloped by the response of the dual foundation system at the majority of the spectral frequencies. An exception to this phenomenon is observed in the horizontal directions, near the fundamental spectral peak, where the "common basemat" spectra exceed the "dual foundation" spectra because of a frequency shift. These exceedances do not have any impact on applying the "dual foundation" results to the design of the RB (with a common basemat) because the "dual foundation" analyses are performed for a multitude of soil cases, and there is always a stiffer soil case (and the fixed base analysis) which accounts for the increase in frequency in the fundamental spectral peak. Therefore, even though the final configuration of the PGC incorporates a common basemat, the "dual foundation" results can still be used for final design purposes.

TABLE 3.7B-1
SEISMIC ANALYSIS CASES

<u>Case</u>	<u>SSE</u>	<u>OBE</u>
Fixed-Base	Yes	Yes
B1	Yes	No
B1.5	Yes	No
B2	Yes	No
B3.5	Yes	Yes
B4	Yes	Yes
C1	Yes	No
C1.5	Yes	No
C2	Yes	No
C3	Yes	No

I

TABLE 3.7B-2
SOIL LAYERS AND PROPERTIES
Case B1

Layer No.	Thick. (ft.)	S-Wave Velocity (ft/sec)	P-Wave Velocity (ft/sec)	Damping S-Wave	Damping P-Wave
1	4.75	1774	4345	0.008	0.003
2	5	1846	4522	0.012	0.004
3	5	1872	4585	0.018	0.006
4	5	1872	4585	0.018	0.006
5	5	1905	4666	0.023	0.008
6	5	1905	4666	0.023	0.008
7	5	1941	4754	0.026	0.009
8	5	1941	4754	0.026	0.009
9	5	1987	4867	0.027	0.009
10	5	1987	4867	0.027	0.009
11	2	1987	4867	0.027	0.009
12	3	2032	4977	0.029	0.010
13	5	2032	4977	0.029	0.010
14	5	2072	5075	0.031	0.010
15	5	2072	5075	0.031	0.010
16	5	2072	5075	0.031	0.010
17	5	2072	5075	0.031	0.010
18	5	2143	5249	0.033	0.011
19	5	2143	5249	0.033	0.011
20	5	2143	5249	0.033	0.011
21	5	2143	5249	0.033	0.011

Unit Weight (all layers) = 125 pcf

TABLE 3.7B-3
SOIL LAYERS AND PROPERTIES

Case B1.5

<u>Layer No.</u>	<u>Thick. (ft.)</u>	<u>S-Wave Velocity (ft/sec)</u>	<u>P-Wave Velocity (ft/sec)</u>	<u>Damping S-Wave</u>	<u>Damping P-Wave</u>
1	4.75	1405	3440	0.010	0.003
2	5	1399	3426	0.019	0.006
3	5	1399	3427	0.027	0.009
4	5	1399	3427	0.027	0.009
5	5	1386	3394	0.036	0.012
6	5	1386	3394	0.036	0.012
7	5	1384	3390	0.041	0.014
8	5	1384	3390	0.041	0.014
9	5	1392	3410	0.045	0.015
10	5	1392	3410	0.045	0.015
11	2	1392	3410	0.045	0.015
12	3	1412	3460	0.047	0.016
13	5	1412	3460	0.047	0.016
14	5	1461	3579	0.047	0.016
15	5	1461	3579	0.047	0.016
16	5	1461	3579	0.047	0.016
17	5	1461	3579	0.047	0.016
18	5	1514	3709	0.049	0.016
19	5	1514	3709	0.049	0.016
20	5	1514	3709	0.049	0.016
21	5	1514	3709	0.049	0.016

Unit Weight (all layers) = 125 pcf

TABLE 3.7B-4
SOIL LAYERS AND PROPERTIES
Case B2

Layer No.	Thick. (ft.)	S-Wave Velocity (ft/sec)	P-Wave Velocity (ft/sec)	Damping S-Wave	Damping P-Wave
1	4.25	996	2440	0.013	0.004
2	4.5	973	2383	0.025	0.008
3	4.5	935	2290	0.039	0.013
4	4.5	935	2290	0.039	0.013
5	4.5	915	2241	0.048	0.016
6	4.5	915	2241	0.048	0.016
7	4.5	915	2241	0.048	0.016
8	4.5	908	2224	0.055	0.018
9	4.5	908	2224	0.055	0.018
10	4.5	909	2227	0.062	0.021
11	4.5	909	2227	0.062	0.021
12	2.5	909	2227	0.062	0.021
13	4.5	909	2227	0.067	0.022
14	4.5	909	2227	0.067	0.022
15	4.5	916	2244	0.072	0.024
16	4.5	916	2244	0.072	0.024
17	4.5	916	2244	0.072	0.024
18	4.5	916	2244	0.072	0.024
19	4.5	945	2315	0.075	0.025
20	4.5	945	2315	0.075	0.025
21	4.5	945	2315	0.075	0.025
22	4.5	945	2315	0.075	0.025
23	3	945	2315	0.075	0.025

Unit Weight (all layers) = 125 pcf

TABLE 3.7B-5**SOIL LAYERS AND PROPERTIES****Case B3.5**

<u>Layer No.</u>	<u>Thick. (ft.)</u>	<u>S-Wave Velocity (ft/sec)</u>	<u>P-Wave Velocity (ft/sec)</u>	<u>Damping S-Wave</u>	<u>Damping P-Wave</u>
1	2.75	567	1389	0.026	0.009
2	3	567	1389	0.026	0.009
3	3	503	1232	0.050	0.017
4	3	503	1232	0.050	0.017
5	3	445	1090	0.080	0.027
6	3	445	1090	0.080	0.027
7	3	445	1090	0.080	0.027
8	3	389	953	0.109	0.036
9	3	389	953	0.109	0.036
10	3	389	953	0.109	0.036
11	3	394	965	0.112	0.037
12	3	394	965	0.112	0.037
13	3	394	965	0.112	0.037
14	3	411	1007	0.110	0.037
15	3	411	1007	0.110	0.037
16	3	411	1007	0.110	0.037
17	2	411	1007	0.110	0.037
18	2	411	1007	0.110	0.037
19	8	1301	3187	0.036	0.012
20	8	1328	3253	0.038	0.013
21	8	1328	3253	0.038	0.013
22	8	1328	3253	0.038	0.013
23	8	1363	3339	0.041	0.014
24	8	1363	3339	0.041	0.014

Unit Weight (all layers) = 125 pcf

TABLE 3.7B-6
SOIL LAYERS AND PROPERTIES

Case B4

<u>Layer No.</u>	<u>Thick. (ft.)</u>	<u>S-Wave Velocity (ft/sec)</u>	<u>P-Wave Velocity (ft/sec)</u>	<u>Damping S-Wave</u>	<u>Damping P-Wave</u>
1	3.75	473	1159	0.025	0.008
2	4	421	1031	0.050	0.017
3	4	421	1031	0.050	0.017
4	3.75	374	916	0.079	0.026
5	3.75	374	916	0.079	0.026
6	3.4	340	833	0.099	0.033
7	3.4	340	833	0.099	0.033
8	3.4	340	833	0.099	0.033
9	3.3	329	806	0.112	0.037
10	3.3	329	806	0.112	0.037
11	3.3	329	806	0.112	0.037
12	3.1	310	759	0.128	0.043
13	3.1	310	759	0.128	0.043
14	3.1	310	759	0.128	0.043
15	3.1	310	759	0.128	0.043
16	8	2124	5203	0.019	0.006
17	10	2181	5342	0.021	0.007
18	10	2181	5342	0.021	0.007
19	10	2264	5546	0.022	0.007
20	10	2264	5546	0.022	0.007

Unit Weight (all layers) = 125 pcf

TABLE 3.7B-7**SOIL LAYERS AND PROPERTIES****Case C1**

<u>Layer No.</u>	<u>Thick. (ft.)</u>	<u>S-Wave Velocity (ft/sec)</u>	<u>P-Wave Velocity (ft/sec)</u>	<u>Damping S-Wave</u>	<u>Damping P-Wave</u>
1	4.75	1770	4335	0.008	0.003
2	7	1844	4518	0.013	0.004
3	8	1866	4571	0.018	0.006
4	8	1901	4656	0.023	0.008
5	8	1937	4744	0.026	0.009
6	8	1937	4744	0.026	0.009
7	8	1976	4839	0.029	0.010
8	8	2001	4902	0.031	0.010
9	10	2046	5011	0.034	0.011
10	10	2046	5011	0.034	0.011
11	10	2138	5236	0.034	0.011
12	10	2138	5236	0.034	0.011
13	10	2310	5658	0.032	0.011
14	10	2310	5658	0.032	0.011
15	10	2266	5550	0.036	0.012
16	10	2266	5550	0.036	0.012
17	10	2359	5778	0.036	0.012
18	10	2359	5778	0.036	0.012
19	10	2450	6001	0.036	0.012
20	10	2450	6001	0.036	0.012
21	10	2424	5938	0.038	0.013
22	10	2424	5938	0.038	0.013

Unit Weight (all layers) = 125 pcf

TABLE 3.7B-8
SOIL LAYERS AND PROPERTIES

Case C1.5

Layer No.	Thick. (ft.)	S-Wave Velocity (ft/sec)	P-Wave Velocity (ft/sec)	Damping S-Wave	Damping P-Wave
1	5.75	997	2443	0.012	0.004
2	6	980	2400	0.024	0.008
3	6	961	2353	0.034	0.011
4	6	961	2353	0.034	0.011
5	6	947	2319	0.043	0.014
6	6	950	2327	0.047	0.016
7	6	950	2327	0.047	0.016
8	6	970	2376	0.049	0.016
9	4	970	2376	0.049	0.016
10	8	976	2391	0.052	0.017
11	8	996	2440	0.056	0.019
12	9	996	2440	0.056	0.019
13	9	1039	2545	0.056	0.019
14	10	1039	2545	0.056	0.019
15	10	1130	2767	0.053	0.018
16	12	1130	2767	0.053	0.018
17	12	1093	2678	0.060	0.020
18	14	1093	2678	0.060	0.020
19	14	1153	2823	0.057	0.019
20	14	1207	2956	0.056	0.019
21	14	1207	2956	0.056	0.019
22	14	1201	2941	0.057	0.019

Unit Weight (all layers) = 125 pcf

TABLE 3.7B-9**SOIL LAYERS AND PROPERTIES****Case C2**

<u>Layer No.</u>	<u>Thick. (ft.)</u>	<u>S-Wave Velocity (ft/sec)</u>	<u>P-Wave Velocity (ft/sec)</u>	<u>Damping S-Wave</u>	<u>Damping P-Wave</u>
1	5.75	485	1188	0.020	0.007
2	6	450	1102	0.039	0.013
3	6	409	1002	0.062	0.021
4	6	375	919	0.085	0.028
5	6	375	919	0.085	0.028
6	6	365	894	0.093	0.031
7	6	365	894	0.093	0.031
8	6	365	894	0.097	0.032
9	4	365	894	0.097	0.032
10	6	369	904	0.100	0.033
11	6	388	950	0.097	0.032
12	6	388	950	0.097	0.032
13	6	388	950	0.097	0.032
14	6	388	950	0.097	0.032
15	7	400	980	0.100	0.033
16	8	400	980	0.100	0.033
17	9	481	1178	0.083	0.028
18	9	481	1178	0.083	0.028
19	9	481	1178	0.083	0.028
20	9	469	1149	0.087	0.029
21	9	469	1149	0.087	0.029
22	9	495	1212	0.085	0.028
23	9	495	1212	0.085	0.028
24	10	515	1261	0.085	0.028
25	10	515	1261	0.085	0.028
26	10	487	1193	0.093	0.031
27	10	487	1193	0.093	0.031

Unit Weight (all layers) = 125 pcf

TABLE 3.7B-10
SOIL LAYERS AND PROPERTIES

Case C3

<u>Layer No.</u>	<u>Thick. (ft.)</u>	<u>S-Wave Velocity (ft/sec)</u>	<u>P-Wave Velocity (ft/sec)</u>	<u>Damping S-Wave</u>	<u>Damping P-Wave</u>
1	4.25	478	1171	0.023	0.008
2	4.5	433	1061	0.045	0.015
3	4.5	386	946	0.073	0.024
4	4.5	386	946	0.073	0.024
5	4.5	386	946	0.073	0.024
6	3.5	357	874	0.092	0.031
7	3.5	357	874	0.092	0.031
8	3.5	354	868	0.097	0.036
9	3.5	354	868	0.097	0.036
10	3.5	354	868	0.097	0.036
11	3.5	348	852	0.107	0.036
12	3.5	348	852	0.107	0.036
13	2.5	348	852	0.107	0.036
14	2.5	348	852	0.107	0.036
15	3.5	357	876	0.107	0.036
16	3.5	357	876	0.107	0.036
17	4	371	908	0.107	0.036
18	4	371	908	0.107	0.036
19	5	371	908	0.107	0.036
20	5	371	908	0.107	0.036
21	6	371	908	0.107	0.036
22	7	394	965	0.104	0.035
23	10	394	965	0.104	0.035
24	25	2500	6123	0.015	0.005
25	25	2461	6028	0.019	0.006
26	25	2539	6220	0.021	0.007
27	25	2626	6433	0.022	0.008

Unit Weight (all layers) = 125 pcf

TABLE 3.7B-11**FREQUENCIES FOR COMPUTATION OF
IN-STRUCTURE RESPONSE SPECTRA (Hz)**

0.1	3.0	9.0	20.0	41.12
0.2	3.15	9.5	20.29	42.0
0.3	3.3	9.79	20.73	43.37
0.4	3.45	10.0	21.0	45.0
0.5	3.6	10.5	22.0	45.78
0.6	3.8	10.75	22.28	46.64
0.7	4.0	11.0	22.55	48.0
0.8	4.2	11.5	22.82	50.0
0.9	4.4	11.79	23.85	50.66
1.0	4.6	12.0	25.0	51.86
1.1	4.8	12.5	25.48	52.52
1.2	5.0	12.76	26.76	53.29
1.3	5.25	13.0	27.33	54.19
1.4	5.5	13.5	27.67	55.64
1.5	5.58	14.0	28.0	57.32
1.6	5.75	14.5	28.48	57.93
1.7	6.0	15.0	28.87	58.96
1.8	6.07	15.23	30.52	60.2
1.9	6.15	15.56	31.0	60.91
2.0	6.25	16.0	31.42	61.94
2.1	6.5	16.79	34.0	64.22
2.2	6.75	17.0	35.0	65.66
2.3	7.0	17.81	36.53	67.56
2.4	7.25	18.0	37.13	70.39
2.5	7.5	18.81	38.0	72.17
2.6	7.75	19.45	38.9	73.48
2.7	8.0	19.7	40.0	75.0
2.8	8.5			78.67
2.9	8.9			100.0

TABLE 3.7B-12

CUTOFF FREQUENCIES IN SSI ANALYSES

<u>Analysis Case</u>	<u>Horizontal Cutoff (Hz)</u>	<u>Vertical Cutoff (Hz)</u>
B1	40	40
B1.5	40	40
B2	40	40
B3.5	33	33
B4	22	36
C1	40	40
C1.5	26	26
C2	12	25
C3	18	18

I

Amendment I
December 21, 1990

SYSTEM 80+™

FLOW DIAGRAM OF SASSI SSI APPROACH
USING FOUNDATION AXISYMMETRY

Figure
3.7B-1

Amendment I
December 21, 1990

SYSTEM 80+™

MODEL OF SOIL PROFILE

Figure

3.7B-2

Amendment I
December 21, 1990

SYSTEM 80 ^{TM} +

AXISYMMETRIC FINITE ELEMENT MESH
FOR THE NEAR-FIELD SOIL AND FOUNDATIONS

Figure
3.7B-3

Amendment I
December 21, 1990

SYSTEM 80 ^{TM} +

AXISYMMETRIC TRANSMITTING BOUNDARIES
FOR IMPEDANCE ANALYSIS

Figure
3.7B-4

Amendment I
December 21, 1990

COMPARISON OF 2% H+V RESPONSE SPECTRA
(SSE, RB FDTN, NODE 131, VERTICAL DIRECTION)

Figure

3.7B-5

Amendment I
December 21, 1990

SYSTEM 80+™

COMPARISON OF 2% H+V RESPONSE SPECTRA
(SSE, RB FDTN, NODE 131, VERTICAL DIRECTION)

Figure
3.7B-6

Amendment I
December 21, 1990

SYSTEM 80 +^{TM}

COMPARISON OF 2% H+V RESPONSE SPECTRA
(SSE, IIS, NODE 210, 0-180 DIRECTION)

Figure

3.7B-7

Amendment I
December 21, 1990

COMPARISON OF 2% H+V RESPONSE SPECTRA
(SSE, IS, NODE 210, VERTICAL DIRECTION)

Figure

3.7B-8

Amendment I
December 21, 1990

SYSTEM 80+™

COMPARISON OF 2% H+V RESPONSE SPECTRA
(SSE, SB, NODE 125, 0-180 DIRECTION)

Figure
3.7B-9

Amendment I
December 21, 1990

SYSTEM 80+™

COMPARISON OF 2% H+V RESPONSE SPECTRA
(SSE, SB, NODE 125, VERTICAL DIRECTION)

Figure

3.7B-10

Amendment I
December 21, 1990

SYSTEM 80+™

COMPARISON OF 2% H+V RESPONSE SPECTRA
(SSE, SCV, NODE 61, VERTICAL DIRECTION)

Figure

3.7B-11

Amendment I
December 21, 1990

SYSTEM 80+™

COMPARISON OF 2% H+V RESPONSE SPECTRA
(SSE, SCV, NODE 61, VERTICAL DIRECTION)

Figure

3.7B-12

SOLID=FIXED-BASE, DASH=84

Amendment I
December 21, 1990

SOLID=FIXED-BASE, DASH=0.5, DOUBLE DASH=0.4

OBE/SSE SPECTRAL RATIO FOR 2% DAMPING
(RB, NODE 131, 90-270 DIRECTION)

Amendment I
December 21, 1990

Figure
3.7B-14

OBE/SSE SPECTRAL RATIO FOR 2% DAMPING
(RB, NODE 131, VERTICAL DIRECTION)

3.7B-15

Figure

Amendment I
December 21, 1990

SOLID=FIXED-BASE, DASH=B3.5, DOUBLE DASH=B4

SOLID=FIXED-BASE, DASH=B3.5, DOUBLE DASH=B4

Amendment I
December 21, 1990

SYSTEM 80 +^{TM}

OBE/SSE SPECTRAL RATIO FOR 2% DAMPING
(IS, NODE 210, 0-180 DIRECTION)

Figure
3.7B-16

SOLID=FIXED-BASE, DASH=B3.5, DOUBLE DASH=B4

OBE/SSE SPECTRAL RATIO FOR 2% DAMPING
(IS, NODE 210, VERTICAL DIRECTION)

Figure
3.7B-18

Amendment I
December 21, 1990

SOLID=FIXED-BASE, DASH=B3.5, DOUBLE DASH=B4

OBE/SSE SPECTRAL RATIO FOR 2% DAMPING
(SB, NODE 125, 0-180 DIRECTION)

Figure
3.7B-19

Amendment I
December 21, 1990

SOLID=FIXED-BASE, DASH=B3.5, DOUBLE DASH=B4

SOLID=FIXED-BASE, DASH=B3.5, DOUBLE DASH=B4

OBE/SSE SPECTRAL RATIO FOR 2% DAMPING
(SB, NODE 125, VERTICAL DIRECTION)

3.7B-21

Figure

Amendment I
December 21, 1990

SOLID=FIXED-BASE, DASH=B3.5, DOUBLE DASH=B4

SOLID=FIXED-BASE, DASH=B3.5, DOUBLE DASH=B4

Amendment I
December 21, 1990

SYSTEM 80 +^{TM}

OBE/SSE SPECTRAL RATIO FOR 2% DAMPING
(SCV, NODE 61, 0-180 DIRECTION)

Figure
3.7B-22

OBE/SSE SPECTRAL RATIO FOR 2% DAMPING
(SCV, NODE 61, 90-270 DIRECTION)

Figure
3.7B-23

Amendment I
December 21, 1990

SOLID=FIXED-BASE, DASH=B3.5, DOUBLE DASH=B4

SOLID=FIXED-BASE, DASH=B3.5, DOUBLE DASH=B4

Amendment I
December 21, 1990

SYSTEM 80 ^{TM} +

SCHEMATIC REPRESENTATION OF PGC MODEL
WITH COMMON BASEMAT

Figure
3.7B-25

COMPARISON OF 2 & 5% RESPONSE SPECTRA
USING COMMON BASEMAT AND DUAL FOUNDATION MODELS
(SSE, RB FDN, NODE 131, 0-180 DIRECTION)

Figure
3.7B-26

Amendment I
December 21, 1988

DASHED=COM. BSMT., SOLID=DUAL FNDTN.

Amendment I
December 21, 1990

SYSTEM 80 +^{TM}

COMPARISON OF 2 & 5% RESPONSE SPECTRA
USING COMMON BASEMAT AND DUAL FOUNDATION MODELS
(SSE, FDTN, NODE 131, VERTICAL DIRECTION)

Figure
3.7B-27

DASHED=COMMON BASMT. SOLID=DUAL FOUNDN.

Amendment I
December 21, 1995

SYSTEM 80+™

COMPARISON OF 2 & 5% RESPONSE SPECTRA
USING COMMON BASEMAT AND DUAL FOUNDATION MODELS
(SSE, IS, NODE 210, 0-180 DIRECTION)

Figure
3.7B-28

COMPARISON OF 2 & 5% RESPONSE SPECTRA
USING COMMON BASEMAT AND DUAL FOUNDATION MODELS
(SSE, IS, NODE 210, VERTICAL DIRECTION)

Figure
3.7B-29

Amendment I
December 21, 1990

DASHED=COM. BSMT., SOLID=DUAL FNDTN.

DASHED=COM. BSMT., SOLID=DUAL FNDTN.

Amendment I
December 21, 1990

SYSTEM 80 +^{TM}

COMPARISON OF 2 & 5% RESPONSE SPECTRA
USING COMMON BASEMAT AND DUAL FOUNDATION MODELS
(SSE, SB, NODE 125, 0-180 DIRECTION)

Figure
3.7B-30

DASHED=COM. BASMT., SOLID=DUAL FNDTN.

Amendment I
December 21, 1990

SYSTEM 80+™

COMPARISON OF 2 & 5% RESPONSE SPECTRA
USING COMMON BASEMAT AND DUAL FOUNDATION MODELS
(SSE, SB, NODE 125, VERTICAL DIRECTION)

Figure
3.7B-31

DASHED=COM. BSMT., SOLID=DUAL FNDTN.

Amendment I
December 21, 1990

SYSTEM 80+™

COMPARISON OF 2 & 5% RESPONSE SPECTRA
USING COMMON BASEMAT AND DUAL FOUNDATION MODELS
(SSE, SCV, NODE 61, 0-180 DIRECTION)

Figure
3.7B-32

DASHED=COM. BSMT., SOLID=DUAL FNDTN.

Amendment I
December 21, 1990

COMPARISON OF 2 & 5% RESPONSE SPECTRA
USING COMMON BASEMAT AND DUAL FOUNDATION MODELS
(SSE, SCV, NODE 61, VERTICAL DIRECTION)

Figure
3.7B-33

3.8   DESIGN OF CATEGORY I STRUCTURES**3.8.1   CONCRETE CONTAINMENT**

This section is not applicable to the System 80+ Standard Design. For a description of the containment, see Section 3.8.2. For a description of the containment shield building, see Section 3.8.4.

3.8.2   STEEL CONTAINMENT**3.8.2.1   Description of the Containment****3.8.2.1.1   General**

The containment is a spherical free-standing welded steel structure. The sphere is supported by sandwiching its lower portion between the building foundation concrete and the interior structure base. There is no structural connection either between the containment and the interior structure, or between the containment and the shield building. The diameter of containment is 200 ft. The plate nominal thickness is 1.75 inches. The containment is shown on the plans and elevations of Figures 1.2-2, 1.2-3, 1.2-5, 1.2-6, 1.2-7 and 1.2-9. I

The spherical shell plate segments will be shop fabricated and field welded. These plates will be approximately 25 feet long and 13 feet wide and can weigh as much as ten tons each; however, these dimensions will vary depending upon the plate location. Two or more plates may be assembled and field welded on the ground and then erected. A vast majority of penetration assemblies will be shop welded to the vessel plates, while others will be attached to vessel in the field. Vessel plate will be thickened around the penetration to compensate for the openings. Where there is a cluster of penetrations in the same plate segment, the entire segment may be fabricated out of the thicker plate, tapered to 1.75 inches at the edges. The additional thickness will depend upon the nominal size, thickness and location of the penetration sleeve and shall be in accordance with ASME Boiler and Pressure Vessel Code (ASME Code) requirements (Reference 1).

3.8.2.1.2   Anchorage Region

The containment is assumed to behave as an independent, free-standing structure above elevation 91+9. Below elevation 91+9, the vessel is encased between the base slab of the internal structures and the shield building foundation. In the transition region, a compressible material is provided as shown in Figure 3.8-1 to eliminate excessive bearing loads on the concrete as

well as to reduce the secondary stresses in the vessel at this location. No shear connectors are provided between the containment plate and the shield building foundation or base slab of internal structures. The lateral loads due to seismic forces, etc., are transferred to the foundation concrete by friction and bearing. No credit is taken for the reduction of shell stresses due to concrete encasement of the vessel; therefore, the vessel plate thickness in the embedded zone is the same as in the free zone.

Penetration assemblies in the concrete subsphere are surrounded by a compressible material as shown in Figure 3.8-2. This compressible material reduces the shear transfer from the containment vessel to the concrete by the penetration assemblies. The compressible material is designed to permit movement of the vessel plate for all design loadings based on a conservative assumption that the plate is free to move between the two layers of concrete. The weld connection between the containment vessel and the penetration sleeve is designed to withstand the nominal shear required to compress the compressible material. The compressible material has a useful life equal to or greater than the life of the plant.

3.8.2.1.3   Containment Penetrations

3.8.2.1.3.1   Equipment Hatch

The equipment hatch is composed of a cylindrical sleeve in the containment shell and a dished head 22 feet in diameter with mating bolted flanges. The flanged joint has double seals with an annular space for pressurized leak testing.

The equipment hatch is designed, fabricated, tested and stamped in accordance with Section III, Subsection NE of the ASME Boiler and Pressure Vessel Code.

Details of a typical equipment hatch are shown on Figure 3.8-1.

3.8.2.1.3.2   Personnel Locks

Two personnel locks 10 feet in diameter are provided for each unit. Each lock has double doors with an interlocking system to prevent both doors being opened simultaneously. Remote indication is provided to indicate the position of each door. Double seals are provided on each door.

The personnel locks are welded steel subassemblies designed, fabricated, tested, and stamped in accordance with Section III, Subsection NE of the ASME Code.

Details of a typical personnel lock are shown on Figure 3.8-1.

3.8.2.1.3.3 Fuel Transfer Penetration

A fuel transfer penetration is provided for transfer of fuel between the fuel pool and the Containment fuel transfer canal. The fuel transfer penetration is provided with a double gasketed blind flange in the transfer canal and a gate valve in the fuel pool. Expansion bellows are provided to accommodate differential movement between the connecting buildings. The fuel transfer penetration is shown in Figure 3.8-2.

3.8.2.1.3.4 Mechanical Penetrations

Mechanical penetrations are treated as fabricated piping assemblies meeting the requirements of the ASME Code, Section III, Subsection NE, and Subsection NC.

The process line and penetration flued head making up the pressure boundary are consistent with the system piping materials; fabrication, inspection, and analysis requirements are as required by the ASME Code, Section III, Subsection NC. All welds on the process pipe are accessible for inspection in accordance with the ASME Code, Section XI. I

High energy lines and selected engineered safety system and auxiliary lines require the typical "Hot Penetration" assembly shown on Figure 3.8-2 which features an exterior guard pipe for protection of other penetrations in the building annular space. Other lines use the typical "Cold Penetration" assembly also shown in Figure 3.8-2 since a leak into the annular space would not cause a personnel hazard or damage other penetrations in the immediate area.

3.8.2.1.3.5 Electrical Penetrations

Medium voltage electrical penetrations for reactor coolant pump power (shown on Figure 3.8-2) use sealed bushings for conductor seals. The assemblies incorporate dual seals along the axis of each conductor.

Low voltage power, control, and instrumentation cables enter the containment vessel through penetration assemblies which are designed to provide two leak-tight barriers in series with each conductor.

All electrical penetrations are designed to maintain containment integrity for Design Basis Accident conditions, including pressure, temperature, and radiation. Double barriers permit testing of each assembly as required to verify that containment integrity is maintained.

3.8.2.2   Applicable Codes, Standards, and Specifications

The containment is designed, fabricated, constructed, and tested in accordance with the appropriate edition of Subsection NE, Section III, of the ASME Boiler and Pressure Vessel Code, "Class MC Components."

The containment, equipment hatch and personnel locks will receive a code stamp in accordance with the requirements of Subsection NE, Section III of the ASME Code.

3.8.2.3   Loads and Loading Combinations

The loads and loading combinations for the analysis and design of the containment vessel are in accordance with Subsection NE, Section III, of the ASME Code and Regulatory Guide 1.57 (Reference 13). In addition, the specified loads and loading combinations are in accordance with the NRC Standard Review Plan, Section 3.8.2, II.3 (Reference 12). The loads and loading combinations are summarized in Tables 3.8-1 and 3.8-2.

A.   Dead Load and Construction Loads

The dead load includes the weight of the containment shell and all permanent attachments. Construction loads are those loads imposed on the containment shell only during construction. A typical construction load is the shoring load induced by the formwork for the containment shield building concrete dome.

B.   Thermal Loads

The containment vessel is subject to thermal loads during normal operation of the unit. The maximum operating temperature can reach 110°F.

C.   Seismic Loads

The containment vessel is loaded by simultaneous seismic events in two orthogonal horizontal directions and the vertical direction. Seismic loads are described in Section 3.7.

D. External Pressure

A vacuum load can be imposed on the Containment Vessel by an accidental trip of the Containment Spray System during normal unit operation. The design vacuum pressure is 2.0 psig.

E. Design Basis Accident

The Design Basis Accident Loads are based on the peak pressure and temperature developed inside Containment as a result of a rupture in the primary coolant system up to and including a double-ended rupture of the largest pipe (a Loss-of-Coolant-Accident or LOCA) or a main steam line break. The containment vessel design pressure is 49 psig and the design temperature is 290°F. See Chapter 6 for details of the Containment Design Basis Accident.

F. Combustible Gas Loads

The containment vessel is subject to the consequences of uncontrolled hydrogen-oxygen recombination as specified in the Code of Federal Regulations, 10 CFR 50.44.

G. Localized Loads

Penetration loads, piping loads and jet impingement loads are all localized loads applied to the containment vessel. Penetration and piping loads are due to the reactions at penetrations, pipe supports/restraints and other attachments welded to the shell. Jet impingement loads are due to fluid jets caused by the rupture of small diameter piping adjacent to the containment vessel.

3.8.2.4   Design and Analysis Procedures

A. Design Per ASME Code

The containment vessel is designed to satisfy the requirements of Articles NE-3130 and NE-3320 of the ASME Code with regard to pressure loadings. The local areas around openings are reinforced as necessary per Article NE-3330.

B. Static and Seismic Load Analysis

The containment vessel is analyzed to determine all membrane, bending and shear stresses resulting from the specified static and seismic loads. The vessel is idealized

as a three dimensional thin shell using the finite element method of analysis. The stresses and deflections produced in the shell under the applied loads are calculated with the ANSYS computer program (Reference 2). The ANSYS mathematical model used to represent the containment vessel is shown in Figure 3.8-3.

Seismic stresses and deflections are calculated using the response spectrum method. The frequencies of vibration and corresponding mode shapes are determined using the normal mode method. Modal responses are combined as described in Regulatory Guide 1.92 (Reference 15). The appropriate damping level for the applied response spectra is defined in Regulatory Guide 1.61 (Reference 14).

C. Buckling

The critical buckling stresses in the containment vessel are determined by applying the appropriate safety factors and capacity reduction factors to the results of a three-dimensional linear bifurcation analysis using an ANSYS finite element model similar to that constructed for the static and dynamic analyses. These methods are described in Article NE-3222 of the ASME Code and ASME Code Case N-284 (Reference 5). Code Case acceptability is in concurrence with Regulatory Guide 1.84 (Reference 18).

D. Ultimate Capacity

The maximum pressure capacity of the containment vessel is evaluated by a large displacement elastic-plastic nonlinear analysis. The vessel is modeled with axisymmetric shell finite elements using the ANSYS computer program. The ANSYS model is shown in Figure 3.8-4.

The stresses in the containment vessel due to combustible gas loadings are calculated using a static linear elastic analysis. The vessel is represented by a three-dimensional thin shell finite element model with the ANSYS computer program. This model is similar to those used for the seismic analysis and buckling evaluation.

E. Nonaxisymmetric and Localized Loads

The containment is not divided into compartments (see Section 6.2.1.2) so there are no nonaxisymmetric loads applied to the containment vessel during a Design Basis Accident.

Localized loads applied to the containment vessel may be piping support/restraint reactions, reactions from other attachments, jet impingement loads, etc. The ANSYS computer program is used to calculate the local stresses caused by these loads, which are then included in the appropriate loading combination.

F. Computer Programs

ANSYS (Reference 2) is a general purpose finite element program capable of both linear and nonlinear analysis. Loads may be either static or dynamic in nature. The program has a large element library useful in modeling a variety of problems.

3.8.2.5 Structural Acceptance Criteria

A. Allowable Stress Limits

Allowable stresses have been established for each of the loading combinations listed in Table 3.8-2. These limits are in compliance with Subsection NE, Section III of the ASME Code and Section 3.8.2, II.5 of the NRC Standard Review Plan. They are summarized in Table 3.8-3. I

B. Buckling Safety Factors

The factors of safety for buckling loads are 3.0 for Level A and B loads, 2.5 for Level C loads and 2.0 for Level D loads. These values are in agreement with the limits set forth in Article NE-3222 of the ASME Code.

C. Ultimate Capacity

The failure criteria for the ultimate capacity analysis is defined in Section III, Appendix II, Article 1430 of the ASME Code. The collapse load is determined by examining the path of the load-displacement curve at various points on the containment vessel.

The containment vessel must satisfy the allowable limits for Level C loadings shown in Table 3.8-3 when subjected to combustible gas loads.

3.8.2.6   Materials, Quality Control, and Special Construction Techniques

3.8.2.6.1   Materials

The containment vessel materials are in accordance with Article NE-2000 of Subsection NE, "Class MC Components," of the ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components."

The containment plate material is ASME SA537 Class 2. This material is exempt from post-weld heat treatment requirements through thicknesses of 1.75 inches.

Fabrication and erection of the containment vessel are in accordance with Article NE-4000 of Section III of the ASME Code. This includes welding procedures, procedure and operator performance qualifications, post weld heat treatment and tolerances.

Nondestructive examination of welds and materials is in accordance with Article NE-5000 of Section III of the ASME Code.

Coatings used on the containment vessel meet the requirements of ANSI N101.2-1972, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities" (Reference 17). This includes the coating materials, surface preparation and application.

3.8.2.6.2   Quality Control

The general provisions of the overall Quality Assurance program are outlined in Chapter 17. These are supplemented by the special provisions of the ASME Code for quality control as applicable to Class MC Components. The containment vessel is ASME Code stamped. Therefore, the ASME Code requirements for quality control have priority over those outlined in Chapter 17 in case of any conflict.

3.8.2.6.3   Special Construction Techniques

The steel containment vessel may be assembled in sections in an area of the construction yard and then lifted and moved into place with a walker crane. This procedure allows the assembly of the containment to begin when the plates forming the lower hemisphere are delivered to the site. In this manner the containment assembly can proceed on a parallel path with the construction of the concrete subsphere region. The following

options are two of several techniques that can be employed for placing the concrete for the dish pedestal supporting the containment:

- A. The concrete dish except for the top four inches can be placed. The containment vessel then would be placed on the support heads and pressure grouted.
- B. The containment vessel can be placed on the support heads and used as formwork for placing the concrete.

With either option care must be taken to prevent the floating of the containment. This is accomplished by filling the containment with water as the grout/concrete is placed. After the lower containment section has been placed, the construction of the interior structure can begin. The assembly of the containment sections will continue in the yard. As the work on the interior structure continues additional sections of the containment can be lifted into place. After the major equipment is placed in the interior structure the top section of the containment vessel can be set. The completed containment will then be used to support the scaffolding for the concrete dome of the shield building.

3.8.2.7   Testing and In-service Surveillance Requirements

The containment vessel, personnel airlocks and equipment hatch are inspected and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE. Penetrations are pressure tested as required for Subsection NC of the ASME Code.

Periodic leakage rate tests of the containment are conducted to verify leak tightness and integrity. These tests and other in-service inspection requirements are described in Section 6.2.

3.8.3   CONCRETE AND STRUCTURAL STEEL INTERNAL STRUCTURES

3.8.3.1   Description of the Internal Structures

The internal structure is a group of reinforced concrete structures that enclose the reactor vessel and primary system. The internal structure provides biological shielding for the containment interior. The internal structure concrete base rests inside the lower portion of the containment vessel sphere. A description of various structures that constitute the internal structure is given in the following paragraphs. The details of the internal structure are shown in Figures 1.2-2, 1.2-3, 1.2-6, 1.2-7 and 1.2-9.

The primary shield wall encloses the reactor vessel and provides protection for the vessel from internal missiles. The primary shield wall provides biological shielding and is designed to withstand the temperatures and pressures following LOCA. In addition, the primary shield wall provides structural support for the reactor vessel. The primary shield wall is a minimum of six feet thick.

The secondary shield wall (crane wall) provides supports for the polar crane and protects the steel containment vessel from internal missiles. In addition to providing biological shielding for the coolant loop and equipment, the crane wall also provides structural support for pipe supports/restraints and platforms at various levels. The crane wall is a right cylinder with an inside diameter of 130 feet and a height of 118 feet from its base. The crane wall is a minimum of four feet thick.

The refueling cavity, when filled with borated water, facilitates the fuel handling operation without exceeding the acceptable level of radiation inside the containment. The refueling cavity has the following sub-compartments:

- A. Storage area for upper guide structure.
- B. Storage area for core support barrel.
- C. Refueling canal.

The refueling canal, when filled with borated water, forms a pool above the reactor vessel. The reactor vessel flange is permanently sealed to the bottom of the refueling canal to prevent leakage of refueling water into the reactor cavity. The fuel transfer tube connects the refueling canal to the Spent Fuel Pool. The refueling canal is filled with borated water to a depth that limits the radiation at the surface of the water to acceptable levels during the period when a fuel assembly is being transferred to the Spent Fuel Pool. The shield walls that form the refueling cavity are a minimum of six feet thick.

The In-containment Refueling Water Storage Tank (IRWST) provides storage of refueling water, a single source of water for the safety injection and containment spray pumps and a heat sink for the Safety Depressurization System. The IRWST is dishlike in shape and utilizes the lower section of the Internal Structure as its outer boundary. The IRWST is provided with a stainless steel liner to prevent leakage. A full description of the IRWST is provided in Section 6.3.2.

The operating floor provides access for operating personnel functions and provides biological shielding. Inside the crane wall, the operating floor is a reinforced concrete slab with a covered hatch that is aligned with hatches in the two lower floors. Outside the crane wall, the operating floor consists of steel grating. There are also reinforced concrete floor slabs at elevation 115+6 and elevation 91+9 that connect the crane wall and the primary shield wall.

The support systems for the reactor vessel, steam generators, reactor coolant pumps and primary loop piping are completely described in Section 5.4.14.

The locations of the missile shield, hatch covers, and other removable structures are shown in Figures 1.2-2, 1.2-3, 1.2-6, 1.2-7 and 1.2-9. The removable slabs and hatch covers are provided with suitable tiedown devices to eliminate any possibility of these items becoming missiles in case of a seismic event or other loading conditions.

3.8.3.2   Applicable Codes, Standards, and Specifications

Category I structures are designed in accordance with the codes and criteria listed in Table 3.8-4.

3.8.3.3   Loads and Loading Combinations

The loads and loading combinations used for the internal structures are shown in Table 3.8-5.

The internal structures are designed for the following loads:

- A. Dead load
- B. Equipment operating loads and other live loads
- C. Pipe reactions
- D. Seismic (See Section 3.7 for seismic criteria)
- E. Internal missiles (The internal structure is designed to withstand internal missiles as defined in Section 3.5.)
- F. Pipe rupture jet impingement
- G. Differential pressures between the reactor vessel cavity, pressurizer enclosure or In-containment Refueling Water Storage Tank and the remainder of the containment free volume.

3.8.3.4   Design and Analysis Procedures

The internal structure is designed for the loads and load combinations specified in Section 3.8.3.3. The complete internal structure (and supporting substructure) is modeled with three-dimensional solid, plate or shell and beam finite elements using ANSYS or another suitable computer code. The forces and moments resulting from the applied static and dynamic loads are used to design the walls, slabs, beams and columns which make up the Internal Structure. The design is performed using either ACI 349-85 (Reference 3) or ANSI/AISC N690-1984 (Reference 4) as appropriate.

3.8.3.5   Structural Acceptance Criteria

The structural acceptance criteria for the Internal Structures is outlined in Section 3.8.4.5.

3.8.3.6   Materials, Quality Control, and Special Construction Techniques

The materials, quality control, and special construction techniques for the concrete internal structures are outlined in Section 3.8.4.6.

3.8.3.7   Testing and In-service Surveillance Requirements

Testing and in-service surveillance requirements are outlined in Section 3.8.4.7.

3.8.4   OTHER CATEGORY I STRUCTURES**3.8.4.1   Description of the Structures****3.8.4.1.1   Containment Shield Building**

The containment shield building is a reinforced concrete structure composed of a right cylinder with a hemispherical dome. The containment shield building shares a common foundation base mat with the nuclear system annex. The containment shield building houses the steel containment vessel and safety-related equipment, and is designed to provide biological shielding as well as external missile protection for the steel containment shell and safety-related equipment.

The containment shield building has an inner radius of 105 feet, a cylinder thickness of 3 feet and a dome thickness of 3 feet. The height of the containment shield building is approximately 215 feet. The structural outline of the containment shield building is shown on Figures 1.2-2 and 1.2-3. An annular space is provided between the containment vessel and containment shield building above elevation 91+9 for structural separation and access to penetrations for testing and inspection.

The reactor building lower volume is that portion of the reactor building which is below elevation 91+9 and external to the containment vessel. The reactor building lower volume houses auxiliary safety-related equipment. This area below the spherical containment allows efficient use of space for locating safety equipment adjacent to the containment vessel and eliminating excessive piping while allowing maximum access to the containment for locating penetrations.

3.8.4.1.2 Nuclear System Annex

The nuclear system annex, as discussed here, includes the Control Building, Diesel Building, Fuel Building, and Main Steam Line Valvehouse. Expansion and contraction joints are used to segment the different areas of the nuclear system annex for structural purposes. Expansion and contraction joints are also used to isolate these areas from adjacent buildings.

The nuclear system annex is a reinforced concrete structure composed of rectangular walls, columns, beams, and floor slabs. The nuclear system annex shares a common foundation base mat with the containment shield building. In addition to the structural components, there are components designed to provide biological shielding and protection against tornado and turbine missiles. Structural components, as well as members serving as shielding components, vary in thickness from approximately one foot to five feet. Details of the nuclear system annex are shown in Figures 1.2-2 through 1.2-10.

3.8.4.1.3 Station Service Water System Structure

There is a Station Service Water System structure which is Category I; however, it is entirely site-specific.

3.8.4.2 Applicable Codes, Standards, and Specifications

Category I structures are designed in accordance with the codes and criteria shown in Table 3.8-4.

3.8.4.3   Loads and Loading Combinations

The Category I structures are designed to maintain their function for the following loadings:

A.   Dead Loads

The dead loads include all sustained loads during and after construction.

B.   Operating Loads

Operating loads are those loads associated with the operation of the plant.

C.   Design Basis Accident Loads

The Design Basis Accident Loads are those associated with the pressure increase in the annulus due to a temperature rise as a result of the energy release inside the containment vessel during a loss-of-coolant accident.

D.   Wind Loads

The wind load is based upon ANSI A58.1-1982 (Reference 6) and ASCE Papers 3269 and 4933 (References 7 and 8) using 130 mph as the fastest mile of wind for a 100 year recurrence period as defined in Section 3.3.1.

The normal and tornado wind loads considered in the design of the containment shield building are nonaxisymmetric loads. The wind loads are analyzed by approximating the wind distribution on the containment shield building as defined in ASCE Paper 4933 by a Fourier Series. The wind distribution curves used in the design are given in Section 3.3.1. Individual harmonics are analyzed and combined to produce the force and moment resultants for the total series.

The wind loads on the Category I structures other than the shield building are analyzed using the methods defined in ASCE Paper 3269.

E.   Tornado Loadings

The tornado loadings are described in Section 3.3.2.

F. Snow and Ice Loads

The containment shield building and the nuclear system annex are designed for a snow and ice load of 50 pounds per square foot.

G. Soil and Water Pressure

The containment shield building and the nuclear system annex are designed for the earth pressure and groundwater pressure defined in Section 2.4.

H. Seismic Loads

See Section 3.7, "Seismic Design," for the seismic loadings.

Loading combinations used for the design of Category I structures are shown in Table 3.8-5.

3.8.4.4   Design and Analysis Procedures

The containment shield building is designed for the static and dynamic loads listed in Section 3.8.4.3. The shield building is modeled as a three dimensional finite element structure with ANSYS or another suitable computer program. The forces and moments determined by the analysis of the applied loads and loading combinations are used for the design of the structure in accordance with ACI 349-85.

The reactor building (including the steel containment vessel, internal structure and containment shield building) is designed to prevent possible overturning, sliding and flotation. The forces and moments acting on the building which could cause these events are determined for the different loads and load combinations and are then compared to the corresponding forces and moments which resist overturning, sliding or flotation. Safety factors for the possible events are determined for comparison with the allowable safety factors listed in Table 3.8-5.

The nuclear system annex is modeled with beam and plate finite elements. It is analyzed for the loads and load combinations found in Section 3.8.4.3. The forces and moments calculated in the analysis are used to design the walls, slabs, beams and columns using either ACI 349-85 or ANSI/AISC N690-1984 as required.

3.8.4.5 Structural Acceptance Criteria

Analysis and design of the concrete internal structures, the shield building, sub-structures, the nuclear system annex and foundations use the ultimate strength design method in accordance with ACI 349-85.

Category I structural steel analysis and design are in accordance with ANSI/AISC N690-1984.

The concrete support for the spherical containment vessel is also analyzed and designed per ACI 349-85. (Since the support has no pressure retaining function, it is not designed in accordance with the ASME Code.)

No increase in allowable stresses for concrete or steel is permitted due to seismic or wind loadings (per the NRC Standard Review Plan, Section 3.8.4, Part II.5).

Three orthogonal components of earthquake loads (2 horizontal and 1 vertical) are considered simultaneously.

A minimum additional eccentricity of $\pm 5\%$ of the maximum building dimension at the level under consideration is assumed to account for accidental torsion (per the NRC Standard Review Plan Section 3.7.2, Part II.11).

In addition to satisfying the load combinations for structural adequacy against the design loadings, the load combinations to ensure safety factors against overturning, sliding, and flotation are checked to ensure overall stability. The following events are checked as a minimum:

- A. The reactor building/nuclear system annex overturning about the toe of the foundation supported on soil.
- B. The reactor building/nuclear system annex foundation sliding on soil.
- C. Floating of the entire reactor building/nuclear system annex foundation base mat.
- D. The containment vessel slipping in the lower concrete support dish.
- E. The containment vessel overturning about the edge of the lower concrete support dish.
- F. The interior structure concrete slipping inside the containment vessel.

The safety factors which must be satisfied during any of these events are shown in Table 3.8-5.

3.8.4.6   Material, Quality Control, and Special Construction Techniques

The Category I structures are poured-in-place reinforced concrete structures. The major materials that will be used in the construction are concrete, reinforcing bars and structural steel. A brief description of these materials is given below.

3.8.4.6.1   Material

3.8.4.6.1.1   Concrete

The basic ingredients of concrete are cement, fine aggregates, coarse aggregates, and mixing water. Admixtures will be used if needed.

Cement will be Type 1 and conform to "Standard Specification for Portland Cement," ASTM C150-89. For special circumstances, other approved cements will be used.

Aggregates will conform to "Standard Specification for Concrete Aggregate," ASTM C33-86. I

Water used in mixing concrete will be clean and free from injurious amounts of oils, acids, alkalis, salts, organic materials or other substances that may be deleterious to concrete or steel. A comparison of the proposed mixing water properties will be made with distilled water by performing the following tests:

- A. Soundness, in accordance with "Standard Test Method for Autoclave Expansion of Portland Cement," ASTM C151-84. The results obtained for the proposed mixing water will not exceed those obtained for distilled water by more than ten percent.
- B. Time of setting, in accordance with "Standard Test Method for Time of Setting of Hydraulic Cement by Vicat Needle," ASTM C191-82. The results obtained for the proposed mixing water will be within ten minutes for initial setting time and one hour for final setting time of those obtained for distilled water.

- C. Compressive strength, in accordance with "Standard Test Method for Compressive Strength of Hydraulic Cement Mortars (using 2 in. cube specimens)," ASTM C109-88. The results obtained for the proposed mixing water will not be lower by more than five percent of those obtained for distilled water.

The water used to make ice for concrete pours in hot weather will conform to the requirements for mixing water described above.

Admixtures, if used and as determined by detailed mix design, will conform with the applicable ASTM standard:

- A. Air-entraining admixtures. "Standard Specification for Air-Entraining Admixtures for Concrete," ASTM C260-86.
- B. Water reducing, retarding, and accelerating admixtures. "Standard Specification for Chemical Admixtures for Concrete," ASTM C494-86.
- C. Pozzolanic admixtures. "Standard Specification for Fly Ash and Raw or Calcined Natural Pozzolan for use as a Mineral Admixture in Portland Cement Concrete," ASTM C618-87.
- D. Slag cement. "Standard Specification for Blended Hydraulic Cements," ASTM C595-86.
- E. Plasticizing admixtures. "Standard Specification for Chemical Admixtures for Use in Producing Flowing Concrete," ASTM C1017-85.

The combined chloride content of the admixtures and mixing water will not exceed 250 ppm.

The ingredient materials will be stored in accordance with the detailed recommendations presented in ACI 304-73 (Reference 11).

Concrete mixes will be designed in accordance with ACI 301-84 (Reference 10). The batching, mixing and transporting of concrete will conform to ACI 301-84. The placement of concrete, consisting of preparation before placing, conveying, depositing, protection and bonding will be in accordance with ACI 301-84.

3.8.4.6.1.2 Reinforcing Steel

Reinforcing steel will consist of deformed reinforcing bars conforming to "Standard Specification for Deformed and Plain Billet - Steel Bars for Concrete Reinforcement," ASTM A615-87,

Grade 60. The fabrication of reinforcing bars, including fabrication tolerances, will be in accordance with CRSI "Manual of Standard Practice" MSP-1-86. The placing of reinforcing bars, including spacing of bars, concrete protection of reinforcement, splicing of bars and field tolerances will be in accordance with ACI 349-85.

3.8.4.6.1.3 Structural Steel

The structural steel will essentially consist of low carbon steel shapes, plates and bars conforming to "Standard Specification for Structural Steel," ASTM A36-87. Other structural steels listed in ANSI/AISC N690-1984 may also be used.

Fabrication and erection of structural steel will be in accordance with the requirements of ANSI/AISC N690-1984. The structural connections will be either welded or bolted. Welding will normally be by electric-arc welding process using electrodes conforming to the requirements of ANSI/AISC N690-1984. All welding will be performed by qualified welders using welding procedures that have been qualified in accordance with American Welding Society (AWS) Structural Welding Code 1988 with latest revisions. Non-destructive examination, if required, will be indicated on the drawings by the Engineer by specifying the type and the Code requirements of the examination to be conducted. All bolted connections will be made with high strength bolts conforming to one of the following specifications:

- A. "Specification for High-Strength Bolts for Structural Steel Joints," ASTM A325-88.
- B. "Specification for Heat-Treated Steel Structural Bolts, 150 KSI Tensile Strength," ASTM A490-88.

Other bolts listed in ANSI/AISC N690-1984 may also be used.

3.8.4.6.2 Quality Control

The quality of materials will be controlled by requiring the suppliers to furnish appropriate mill test reports as required under relevant ASTM Specifications as described in Subsection 3.8.4.6.1. These mill test reports will be reviewed and approved in accordance with the general provisions of the overall Quality Assurance Program outlined in Chapter 17 and supplemented by the special provisions of the appropriate codes and specifications for design listed in Table 3.8-4.

Erection tolerances, in general, will be in accordance with the referenced design code. Where special tolerances that influence the erection of equipment, etc., are required, they will be indicated on the drawings by the Engineer.

3.8.4.6.3   Special Construction Techniques

No unique or untried construction techniques are contemplated. Both the cylindrical and the dome portions of the shield building will be constructed using standard construction techniques.

3.8.4.7   Testing and In-service Surveillance Requirements

There will be no testing or in-service surveillance beyond those quality control tests performed during construction, which will be in accordance with ACI 349-85, ACI 301-84, ANSI/AISC N690-1984 or ANSI N45.2.5-1978 (Reference 9) as applicable.

3.8.5   FOUNDATIONS

3.8.5.1   Description of the Foundations

The foundations of the Category I structures are thick reinforced concrete mats. The foundation of the reactor building/nuclear system annex complex is approximately 10 feet thick, has a flat bottom and rests on soil or rock. The foundations of other Category I structures are also reinforced concrete mats on soil or rock.

3.8.5.2   Applicable Codes, Standards, and Specifications

Reinforced concrete foundations and supports of Category I structures are designed in accordance with ACI 349-85.

3.8.5.3   Loads and Loading Combinations

The design loads and loading combinations are described in Section 3.8.4.3.

3.8.5.4   Design and Analysis Procedures

The reinforced concrete foundations of Category I structures are analyzed and designed for the reactions due to static, seismic and all other significant loads at the base of the superstructures supported by the foundation. The foundation mat is modeled as a flat plate on an elastic foundation. The forces and moments determined in the analysis are input to the structural design using ACI 349-85.

The analysis and design of the foundations require site specific soil properties. Therefore, the detailed design of the foundations and the verification of their acceptability for sliding and overturning shall be performed on a site specific basis.

3.8.5.5   Structural Acceptable Criteria

These are outlined in Section 3.8.4.5.

3.8.5.6   Material, Quality Control, and Special Construction Techniques

These are outlined in Section 3.8.4.6.

3.8.5.7   Testing and In-service Surveillance Requirements

These are outlined in Section 3.8.4.7.

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REFERENCES FOR SECTION 3.8

1. 1989 ASME Boiler and Pressure Vessel Code.
2. Gabriel J. DeSalvo and John A. Swanson, ANSYS Engineering Analysis System User's Manual, Swanson Analysis Systems, Inc.
3. ACI 349-85, "Code Requirements for Nuclear Safety Related Concrete Structures".
4. ANSI/AISC N690-1984, "Nuclear Facilities -- Steel Safety-Related Structures for Design Fabrication and Erection".
5. ASME Code Case N-284, "Metal Containment Shell Buckling Design Methods, Section III, Division 1, Class MC".
6. "Minimum Design Loads for Buildings and Other Structures," ANSI A58.1-1982. I
7. "Wind Forces on Structures," ASCE Paper No. 3269, Transactions, ASCE, Vol. 126, Part II, 1961, p. 1124.
8. "Wind Loads on Dome-Cylinder and Dome-Cone Shapes", ASCE Paper No. 4933, Journal of the Structural Division, ASCE, Vol. 92, No. ST5, October 1966, p.79.
9. "Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete, Structural Steel, Soils and Foundations During the Construction Phase of Nuclear Power Plants," ANSI N45.2.5-1978.
10. ACI 301-84, "Specifications for Structural Concrete for Buildings".
11. ACI 304-73, "Recommended Practice for Measuring, Mixing, Transporting and Placing Concrete".
12. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants".
13. Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components".

14. Regulatory Guide 1.61, "Damping Valves for Seismic Design of Nuclear Power Plants".
15. Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis".
16. Code of Federal Regulations, Title 10, Part 50.
17. ANSI N101.2-1972, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities".
18. Regulatory Guide 1.84, Design and Fabrication Code Case Acceptability ASME Section III Division 16 .

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TABLE 3.8-1

(Sheet 1 of 2)

DESIGN LOADINGS FOR STEEL CONTAINMENT

D	---	Dead loads
C	---	Construction loads
L	---	Live loads
P _t	---	Test pressure
T _t	---	Test temperature
T _o	---	Thermal loads during startup, normal operating or shutdown conditions
R _o	---	Pipe reactions during startup, normal operating or shutdown conditions
P _o	---	Pressure loads during startup, normal operating or shutdown conditions
P _e	---	External pressure loads resulting from pressure variation either inside or outside containment
T _e	---	Thermal effects associated with external pressure loads
R _e	---	Pipe reactions associated with external pressure loads
E	---	Loads generated by the operating basis earthquake
E'	---	Loads generated by the safe shutdown earthquake
P _a	---	Pressure load generated by the postulated pipe break accident
T _a	---	Thermal loads generated by the postulated pipe break accident
R _a	---	Pipe reactions due to thermal conditions generated by the postulated pipe break accident
Y _r	---	Equivalent static load on the structure generated by the reaction on the broken pipe during the design basis accident

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TABLE 3.8-1 (Cont'd)

(Sheet 2 of 2)

DESIGN LOADINGS FOR STEEL CONTAINMENT

Y_j	---	Jet impingement equivalent static load on the structure generated by the broken pipe during the design basis accident
Y_m	---	Missile impact equivalent static load on the structure generated by or during the design basis accident, such as pipe whipping
F_L	---	Loads generated by the post-LOCA flooding of the containment

TABLE 3.8-2

(Sheet 1 of 2)

LOADING COMBINATIONS FOR STEEL CONTAINMENT

The containment vessel may be subject to the combined effect of two or more of the design loadings listed in Table 3.8-1. The stresses computed by analysis for each design loading shall be combined in the following manner:

A. Testing Condition:

This includes the testing condition of containment to verify its leak integrity.

$$D + L + P_t + T_t$$

B. Design Conditions:

These include all design loadings to which the containment vessel might be subjected during the expected life of the plant

$$D + L + T_a + R_a + P_a$$

C. Service Conditions:

The loads corresponding to the service limits may be combined by their actual time history of occurrence while considering their dynamic effect upon the structure.

Level A Service Limits

$$\begin{array}{l} D + L + T_o + R_o + P_o \\ D + L + T_a + R_a + P_a \end{array}$$

Level B Service Limits

$$\begin{array}{l} D + L + T_a + R_a + P_a + E \\ D + L + T_o + R_o + P_o + E \end{array}$$

Level C Service Limits:

$$\begin{array}{l} D + L + T_a + R_a + P_a + E' \\ D + L + T_o + R_o + P_o + E' \end{array}$$

TABLE 3.8-2 (Cont'd)**(Sheet 2 of 2)****LOADING COMBINATIONS FOR STEEL CONTAINMENT**

Level D Service Limits:

$$D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + E'$$

D. Post-LOCA Conditions:

$$D + L + F_L + E$$

E. Construction Conditions:

$$D + C$$

F. Stability Considerations:

The following combinations are considered for stability with the containment subject to external pressure.

$$\begin{array}{l} D + L + T_e + R_e + P_e + E \\ D + L + T_e + R_e + P_e \end{array}$$

These combinations are in accordance with the applicable load combinations in Section 3.8.2 of the NRC Standard Review Plan. The load combinations also meet the Code Requirements of Subsection NE, Section III, Division 1, of the ASME Code. The Service Limits are defined in the ASME Code.

TABLE 3.8-3

(Sheet 1 of 3)

STRESS INTENSITY LIMITS FOR STEEL CONTAINMENTS

Load Categories		Primary Stresses Gen. Mem. P_m	Local Mem. P_L	Bending & Local Mem. $P_b + P_L$ (6)	Primary & Secondary $P_L + P_b + Q$	Peak Stresses $P_L + P_b + Q + F$	Buckling
Testing Condition	Pneumatic	$0.75S_y$	$1.15S_y$	$1.15S_y$	N/A (2)	Consider for (5) fatigue evaluation	See (9)
Design Condition		$1.0S_{mc}$	$1.5S_{mc}$	$1.5S_{mc}$	N/A	N/A	See (9)
Level A Service Limit (1)		$1.0S_{mc}$	$1.5S_{mc}$	$1.5S_{mc}$	$3.0S_{mi}$	Consider for fatigue evaluation	See (9)
Level B Service Limit		$1.0S_{mc}$	$1.5S_{mc}$	$1.5S_{mc}$	$3.0S_{mi}$	Consider for fatigue evaluation	See (9)

TABLE 3.8-3 (Cont'd)

(Sheet 2 of 3)

STRESS INTENSITY LIMITS FOR STEEL CONTAINMENTS

Load Categories		Primary Stresses Gen. Mem. P_m	Local Mem. P_L	Bending & Local Mem. $P_b + P_L$ (6)	Primary & Secondary $P_L + P_b + Q$	Peak Stresses $P_L + P_b + Q + F$	Buckling
Level C Service Limit	Not integral and Continuous	$1.0S_{mc}$	$1.5S_{mc}$	$1.5S_{mc}$	$3.0S_{mi}$	N/A	See (9)
	Integral and Continuous (4)(7)	$1.2S_{mc}$ or * $1.0S_y$	$1.8S_{mc}$ or * $1.5S_y$	$1.8S_{mc}$ or * $1.5S_y$ (8)	N/A	N/A	See (9)
Level D Service Limit	Not integral and Continuous	$1.2S_{mc}$ or * $1.0S_y$	$1.8S_{mc}$ or * $1.5S_y$	$1.8S_{mc}$ or * $1.5S_y$ (8)	N/A	N/A	See (9)
	Integ. & Cont.	Elastic An. (3) S_f	$1.5S_f$	$1.5S_f$	N/A	N/A	See (9)
		Inelastic An. (3) S_f	S_f	S_f			
Post-Flooding		$1.2S_{mc}$ or * $1.0S_y$	$1.8S_{mc}$ or * $1.5S_y$	$1.8S_{mc}$ or * $1.5S_y$	$3.0S_{mi}$	N/A (2)	See (9)

TABLE 3.8-3 (Cont'd)

(Sheet 3 of 3)

STRESS INTENSITY LIMITS FOR STEEL CONTAINMENTS

NOTES:

- (1) The allowable stress intensity S_{mi} shall be the S_m listed in Tables I-1.0 and the allowable stress intensity S_{mc} shall be the S_m listed in Tables I-10.0 of Appendix I of the ASME Code.
- (2) N/A - No evaluation required.
- (3) S_f is 85% of the general primary membrane allowable permitted in Appendix F. In the application of the rules of Appendix F, S_{mi} , if applicable, shall be as specified in Tables I-1.0.
- (4) Those limits identified by the asterisk (*) indicate a choice of the larger of the two limits.
- (5) The number of test sequences shall not exceed 10 unless a fatigue evaluation is considered.
- (6) Values shown are for a solid rectangular section. See NE-3220 for other than a solid rectangular section.
- (7) These stress intensity limits apply to the partial penetration welds also.
- (8) Values shown are applicable when $P_L < 0.67S_y$. When $P_L > 0.67S_y$, use the larger of the two limits, $[2.5 - 1.5(P_L/S_y)]1.2S_{mc}$ or $[2.5 - 1.5(P_L/S_y)]S_y$.
- (9) It must be demonstrated that any axisymmetric techniques proposed are applicable to a vessel having large asymmetric openings, and that the overall margin of safety to prevent buckling is adequate.

TABLE 3.8-4

CODES AND SPECIFICATIONS FOR DESIGN OF CATEGORY I STRUCTURES

<u>Structural Component</u>	<u>Design Codes and Specifications</u>
Concrete	ACI 349-85
Concrete Reinforcement	ASTM A615
Structural Steel and Plates	ANSI/AISC N690-1984 ASTM A-36
Containment Vessel Shell	Subsection NE, Section III of the ASME Code

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TABLE 3.8-5

(Sheet 1 of 10)

LOAD COMBINATIONS FOR CATEGORY I STRUCTURES

INDEX

I. Load Definitions

1. Normal Loads
2. Severe Environmental Loads
3. Extreme Environmental Loads
4. Abnormal Loads
5. Other Definitions

II. Load Combinations and Acceptance Criteria for Category I Concrete Structures

1. Service Load Conditions
2. Factored Load Conditions

III. Load Combinations and Acceptance Criteria for Category I Steel Structures

1. Service Load Conditions
  - a. Elastic Design
  - b. Plastic Design
2. Factored Load Conditions
  - a. Elastic Design
  - b. Plastic Design

IV. Load Combinations and Acceptance Criteria for Category I Foundations

TABLE 3.8-5 (Cont'd)**(Sheet 2 of 10)****LOAD COMBINATIONS FOR CATEGORY I STRUCTURES****I. Load Definitions**

All the major loads to be encountered and/or to be postulated in a Category I structure are grouped into four categories described below. All the loads listed, however, are not necessarily applicable to all the structures and their elements in the plant. Loads and the applicable load combinations for which each structure is designed will depend on the conditions to which that particular structure could be subjected.

1. Normal Loads

Normal loads are those loads to be encountered during normal plant operation and shutdown. They include the following:

- D --- Dead loads or their related internal moments and forces, including any permanent equipment loads and hydrostatic loads
- L --- Live loads or their related internal moments and forces, including any movable equipment loads and other loads which vary with intensity and occurrence, such as soil pressure
- F' --- Buoyant force of probable maximum flood
- T₀ --- Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady state condition
- R₀ --- Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady state condition

2. Severe Environmental Loads

Severe environmental loads are those loads that could infrequently be encountered during the plant life. Included in this category are:

- E --- Loads generated by the Operating Basis Earthquake
- W --- Loads generated by the design wind specified for the plant

TABLE 3.8-5 (Cont'd)

(Sheet 3 of 10)

LOAD COMBINATIONS FOR CATEGORY I STRUCTURES3. Extreme Environmental Loads

Extreme environmental loads are those which are credible but are highly improbable. They include:

E' --- Loads generated by the Safe Shutdown Earthquake

W_t --- Loads generated by the Design Basis Tornado specified for the plant. They include loads due to the tornado wind pressure (W_w), loads due to the tornado-created differential pressures (W_p), and loads due to the tornado-generated missiles (W_m)

The combined effect of W_w , W_p , and W_m is determined in a conservative manner for each particular structure or portion thereof, as applicable, by using one or more of the following combinations as appropriate:

(i) $W_t = W_w$

(ii) $W_t = W_p$

(iii) $W_t = W_m$

(iv) $W_t = W_w + 0.5 W_p$

(v) $W_t = W_w + W_m$

(vi) $W_t = W_w + 0.5 W_p + W_m$

4. Abnormal Loads

Abnormal loads are those loads generated by a postulated high energy pipe break accident within a building and/or compartment thereof. Included in this category are the following:

P_a --- Pressure equivalent static load within or across a compartment and/or building, generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load

TABLE 3.8-5 (Cont'd)**(Sheet 4 of 10)****LOAD COMBINATIONS FOR CATEGORY I STRUCTURES**

- T_a --- Thermal loads under thermal conditions generated by the postulated break and including T_o
- R_a --- Pipe reactions under thermal conditions generated by the postulated break and including R_o
- Y_r --- Equivalent static load on the structure generated by the reaction on the broken high-energy pipe during the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load
- Y_j --- Jet impingement equivalent static load on a structure generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load
- Y_m --- Missile impact equivalent static load on a structure generated by or during the postulated break, such as pipe whipping, and including an appropriate dynamic load factor to account for the dynamic nature of the load

In determining an appropriate equivalent static load for Y_r , Y_j , and Y_m , elastic-plastic behavior may be assumed with appropriate ductility ratios as long as excessive deflections will not result in loss of function of any safety related system.

5. Other Definitions

- S --- For structural steel, S is the required section strength based on the elastic design methods and the allowable stresses defined in ANSI/AISC N690-1984
- U --- For concrete structures, U is the section strength required to resist design loads based on the ultimate strength design method described in ACI 349-85
- Y --- For structural steel, Y is the section strength required to resist design loads based on plastic design methods described in ANSI/AISC N690-1984

TABLE 3.8-5 (Cont'd)**(Sheet 5 of 10)****LOAD COMBINATIONS FOR CATEGORY I STRUCTURES****II. Load Combinations and Acceptance Criteria for Category I Concrete Structures**

The following set of load combinations and allowable design limits is used for all Category I concrete structures:

1. Service Load Conditions

Service Load Conditions, represent Normal, Severe Environmental and Normal/Severe Environmental loads.

The Ultimate Strength Design method is used with the following load combinations:

- 1) $U = 1.4 D + 1.7 L$
- 2) $U = 1.4 D + 1.7 L + 1.9 E$
- 3) $U = 1.4 D + 1.7 L + 1.7 W$

If thermal stresses due to T_o and R_o are present, the following combinations are also satisfied:

- 4) $U = (0.75) (1.4 D + 1.7 L + 1.7 T_o + 1.7 R_o)$
- 5) $U = (0.75) (1.4 D + 1.7 L + 1.9 E + 1.7 T_o + 1.7 R_o)$
- 6) $U = (0.75) (1.4 D + 1.7 L + 1.7 W + 1.7 T_o + 1.7 R_o)$

Both cases of L having its full value or being completely absent are checked.

In addition the following combinations are considered:

- 7) $U = 1.2 D + 1.9 E$
- 8) $U = 1.2 D + 1.7 W$

Where soil and/or hydrostatic pressures are present, in addition to all the above combinations where they have been included in L and D , respectively, the requirements of ACI 349-85 are also satisfied.

TABLE 3.8-5 (Cont'd)**(Sheet 6 of 10)****LOAD COMBINATIONS FOR CATEGORY I STRUCTURES****2. Factored Load Conditions**

Factor Load Conditions represent Extreme Environmental, Abnormal, Abnormal/Severe Environmental and Abnormal/Extreme Environmental loads. The Ultimate Strength Design method is used with the following load combinations:

- 1) $U = D + L + T_o + R_o + E'$
- 2) $U = D + L + T_o + R_o + W_t$
- 3) $U = D + L + T_a + R_a + 1.5 P_a$
- 4) $U = D + L + T_a + R_a + 1.25 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.25 E$
- 5) $U = D + L + T_a + R_a + 1.0 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.0 E'$

In factored load combinations (3), (4), and (5), the maximum values of P_a , T_a , R_a , Y_j , Y_r , and Y_m , including an appropriate dynamic load factor, are used unless a time-history analysis is performed to justify otherwise. Factored load combinations (2), (4), and (5) are satisfied first without the tornado missile load in (2), and without Y_r , Y_j , and Y_m in (4) and (5). When considering these loads, however, local section strength capacities may be exceeded under the effect of 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 are checked.

Where any load reduces the effects of other loads, the corresponding coefficient for that load should be taken as 0.9 if it can be demonstrated that the load is always present or occurs simultaneously with other loads. Otherwise the coefficient for the load should be taken as zero.

Where the structural effects of differential settlement, creep, or shrinkage may be significant, they should be included with the dead load, D , as applicable.

TABLE 3.8-5 (Cont'd)**(Sheet 7 of 10)****LOAD COMBINATIONS FOR CATEGORY I STRUCTURES****III. Load Combinations and Acceptance Criteria for Category I Steel Structures**

The following set of load combinations and allowable design limits is used for all Category I steel structures:

1. Service Load Conditions

Either the elastic working stress design methods or the plastic design methods of ANSI/AISC N690-1984 may be used.

a. If the elastic working stress design methods are used:

1) $S = D + L$

2) $S = D + L + E$

3) $S = D + L + W$

If thermal stresses due to T_o and R_o are present, the following combinations are also satisfied:

4) $1.5 S = D + L + T_o + R_o$

5) $1.5 S = D + L + T_o + R_o + E$

6) $1.5 S = D + L + T_o + R_o + W$

Both cases of L having its full value or being completely absent are checked.

b. If plastic design methods are used:

1) $Y = 1.7 D + 1.7 L$

2) $Y = 1.7 D + 1.7 L + 1.7 E$

3) $Y = 1.7 D + 1.7 L + 1.7 W$

TABLE 3.8-5 (Cont'd)**(Sheet 8 of 10)****LOAD COMBINATIONS FOR CATEGORY I STRUCTURES**

If thermal stresses due to T_o and R_o are present, the following combinations are also satisfied:

$$4) \quad Y = 1.3(D + L + T_o + R_o)$$

$$5) \quad Y = 1.3(D + L + E + T_o + R_o)$$

$$6) \quad Y = 1.3(D + L + W + T_o + R_o)$$

Both cases of L having its full value or being completely absent are checked.

2. Factored Load Conditions

The following load combinations are satisfied:

a. If elastic working stress design methods are used:

$$1) \quad 1.6 S = D + L + T_o + R_o + E'$$

$$2) \quad 1.6 S = D + L + T_o + R_o + W_t$$

$$3) \quad 1.6 S = D + L + T_a + R_a + P_a$$

$$4) \quad 1.6 S^* = D + L + T_a + R_a + P_a + 1.0 (Y_j + Y_r + Y_m) + 1.0 E$$

$$5) \quad 1.7 S^* = D + L + T_a + R_a + P_a + 1.0 (Y_j + Y_r + Y_m) + 1.0 E'$$

* For these two combinations, (4) and (5), in computing the required section strength, S , the plastic section modulus of steel shapes may be used.

b. If plastic design methods are used:

$$1) \quad Y^* = D + L + T_o + R_o + E'$$

$$2) \quad Y^* = D + L + T_o + R_o + W_t$$

$$3) \quad Y^* = D + L + T_a + R_a + 1.5 P_a$$

$$4) \quad Y^* = D + L + T_a + R_a + 1.25 P_a + 1.0 (Y_j + Y_r + Y_m) + 1.25 E$$

$$5) \quad Y^* = D + L + T_a + R_a + 1.0 P_a + 1.0 (Y_j + Y_r + Y_m) + 1.0 E'$$

TABLE 3.8-5 (Cont'd)

(Sheet 9 of 10)

LOAD COMBINATIONS FOR CATEGORY I STRUCTURES

In the above factored load combinations, thermal loads can be neglected when it can be shown that they are secondary and self-limiting in nature and where the material is ductile.

* Y (for the factored load combinations) should be multiplied by 0.90 for the Internal Structures and 1.0 for other Category I structures.

In factored load combinations (3), (4), and (5), the minimum values of P , T , T_o , Y_j , Y_r , and Y_m , including an appropriate dynamic load factor, are used unless a time-history analysis is performed to justify otherwise. Factored load combinations (2), (4), and (5) are first satisfied without the tornado missile load in (2), and without Y_r , Y_j , and Y_m in (4) and (5). When considering these loads, however, local section strengths may be exceeded under the effect of these concentrated loads, provided there will be no loss of function of any safety-related system.

Where any load reduces the effects of other loads, the corresponding coefficient for that load should be taken as 0.9, if it can be demonstrated that the load is always present or occurs simultaneously with other loads. Otherwise, the coefficient for that load should be taken as zero.

Where the structural effect of differential settlement may be significant it should be included with the dead load, D .

IV. Load Combinations and Acceptance Criteria for Category I Foundations

In addition to the load combinations and acceptance criteria referenced above, all Category I foundations are also checked against sliding and overturning due to earthquakes, winds, and tornadoes and against flotation due to floods in accordance with the following:

<u>Load Combination</u>	<u>Minimum Factors of Safety</u>		
	<u>Overturning</u>	<u>Sliding</u>	<u>Flotation</u>
$D + H + E$	1.5	1.5	-
$D + H + W$	1.5	1.5	-
$D + H + E'$	1.1	1.1	-
$D + H + W_t$	1.1	1.1	-
$D + F'$	-	-	1.1

TABLE 3.8-5 (Cont'd)

(Sheet 10 of 10)

LOAD COMBINATIONS FOR CATEGORY I STRUCTURES

Definitions:

D	-	Dead Load
F'	-	Buoyant Forces of Design Basis Flood
H	-	Lateral Earth Pressure
E	-	OBE Seismic Load
E'	-	SSE Seismic Load
W	-	Wind Load
W _t	-	Tornado Load

I

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Figure
3.8-1

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Figure

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Figure
3.8-1

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**CATEGORY I STRUCTURES
TYPICAL HOT PENETRATION**

Figure

3.8-2

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**CATEGORY I STRUCTURES
TYPICAL COLD PENETRATION**

**Figure
3.8-2**

Figure 3.8-2	CATEGORY I STRUCTURES TYPICAL SMALL PENETRATION	

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**CATEGORY I STRUCTURES
TYPICAL SUBSPHERE PENETRATION**

Figure

3.8-2

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**CATEGORY I STRUCTURES
TYPICAL ELECTRICAL PENETRATION**

Figure

3.8-2

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**CATEGORY I STRUCTURES
TYPICAL FUEL TRANSFER PENETRATION**

Figure

3.8-2

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THREE-DIMENSIONAL ANSYS CONTAINMENT MODEL

Figure
3.8-3

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AXISYMMETRIC ANSYS CONTAINMENT MODEL

Figure

3.8-4

END

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