3.0 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

Because of numerous, multiple cross-references to other WAPWR modules in this section, a list of modules and their appropriate title is provided here for convenience. The module title will be included in the text where a single module is referenced.

WAPWR Module	WAPWR Module Title
1	Primary Side Safeguards System
2	Regulatory Conformance
3	Introduction and Site
4	Reactor Coolant System
5	Reactor System
6	Secondary Side Safeguards System (*)
7	Structural/Equipment Design
8	Steam and Power Conversion (*)
9	I&C Electrical Power
10	Containment Systems
11	Radiation Protection
12	Waste Management
13	Auxiliary Systems
14	Initial Test Program
15	ACR/Human Factors
16	PRA/Severe Accident
17	Completed Application

(*) Modules merged into one module.

This chapter identifies, describes, and discusses the principal architectural and engineering design features of those structures, components, equipment, and systems that are necessary to assure:

A. The integrity of the reactor coolant pressure boundary.

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

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

3.1 Conformance with Nuclear Regulatory Commission (NRC) General Design Criteria(GDC)

This section discusses briefly the extent to which the design criteria for structures, systems, and components important to safety comply with Title 10, Code of Federal Regulations, Part 50 (10CFR50), Appendix A, "General Design Criteria for Nuclear Power Plants". As presented in this section, each criterion is first quoted and then discussed in sufficient detail to demonstrate compliance. For some criteria, where additional information may be required for a complete discussion, detailed evaluations of compliance with each criterion are incorporated in more appropriate sections, but are located by reference.

3.1.1 Overall Requirements

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 function 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."

Discussion

The quality assurance program for the WAPWR, together with the quality assurance, quality engineering, and quality control programs of the major contractors and their vendors, ensure that structures, systems, and components important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the safety functions to be performed. This is accomplished through the use of recognized codes, standards, and design criteria. As necessary, additional supplemental standards, design criteria, and requirements are developed by Westinghouse and the major contractors' engineering organizations. Appropriate records associated with the engineering and design, fabrication, erection, and testing which document the compliance with recognized codes, standards, and design criteria are maintained throughout the life of the units either by or under the control of the applicant. Quality assurance is described in Chapter 17.0 of RESAR-SP/90 PDA Modules 1, 3 through 14, and in Chapter 17.0 of the integrated RESAR-SP/90 PDA document.

The principal design criteria, design bases, codes, and standards applied to the facility are described in Section 3.2. Additional detail may be found in the pertinent section of the document dealing with structures, systems, and components important to safety, e.g., the containment as described in Subsection 3.8.1 of this module.

Criterion 2 - Design Bases for Protection Against Natural Phenomena

"Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without the loss of the 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."

Discussion

The structures, systems, and components important to safety are designed either to withstand the effects of natural phenomena without loss of the capability to perform their safety functions, or are designed such that their response or failure will be in a safe condition. Those structures, systems, and components vital to the shutdown capability of the reactor are designed to withstand the maximum probable natural phenomena at the site, determined from recorded data for the site vicinity, with appropriate margins to account for uncertainties in historical data. Appropriate combinations of structural loadings from normal, accident, and natural phenomena are considered in the plant design. The design of the plant in relationship to those natural events is addressed throughout this module. Seismic and quality group classifications, as well as other pertinent standards and information, are given in the sections discussing individual structures and components and in Table 3.2-1. The nature and magnitude of the natural phenomena considered in the design of this plant are discussed in Chapter 2.0 of RESAR-SP/90 PDA Module 3, "Introduction and Site".

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."

Discussion

The plant is designed to minimize the probability and effect of fires and explosions. Noncombustible and fire-resistant materials are used in the containment, control room, components of safety systems, and throughout the unit wherever fire is a potential risk to safety-related systems. For example, electrical cables have a fire-retardant jacketing, and fire barriers and fire stops are utilized as described in Subsection 9.5.1 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems". Equipment and facilities for fire protection, including detection, alarm, and extinguishment, are provided to protect both plant equipment and personnel from fire, explosion, and the resultant release of toxic vapors. Fire protection is provided by deluge systems (water spray), sprinklers, Halon 1301, and portable extinguishers. Firefighting systems are designed to assure that their rupture or inadvertent operation will not prevent systems important to safety from performing their design functions.

The following codes, guides, and standards are used as guidelines in the design of the fire protection system and equipment. The system and equipment substantially conform to the applicable portions of the following documents:

A. National Fire Protection Association (NFPA) "National Fire Codes."

B. BTP-CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," July 1981.

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

Discussion

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 LOCAs. Criteria are presented in Chapter 3, and the environmental conditions are described in Section 3.11.

These structures, systems, and components are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. Details of the design, environmental testing, and construction of these systems, structures, and components are included in Chapters 3, 5, 6, 7, 9, and 10 in RESAR-SP/90 PDA (Modules 1, 4, 6, 7, 8, 9, 10 & 13). Evaluation of the performance of the safety features is contained in Chapter 15 of RESAR-SP/90 PDA (Modules 1, 4, 5, 6, 8, 10, 12, 13 & 16).

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 unit."

Discussion

The WAPWR is a single unit plant.

3.1.2 Protection by Multiple Fission Product Barriers

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."

Discussion

The reactor core and associated coolant, control, and protection systems are designed to the following criteria:

A. No fuel damage will occur during normal core operation and operational transients (Condition 1) or any transient conditions arising from occurrences of moderate frequency (Condition 2) beyond a small fraction of clad defects for which various aspects of the plant are designed. Fuel damage, as used here, is defined as penetration of the fission product barrier, i.e., the fuel rod clad. The small number of clad defects that may occur are within the capability of the plant cleanup system and are consistent with the plant design bases.

B. The reactor can be returned to a safe shutdown state following a Condition 3 event with only a small fraction of the fuel rods damaged, although sufficient fuel damage might occur to preclude the immediate resumption of operation.

C. The core will remain intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition 4).

The reactor trip system is designed to actuate a reactor trip whenever necessary to ensure that the fuel design limits are not exceeded. The core design, together with the process and decay heat removal systems, provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of the loss of reactor coolant flow, trip of the turbine generator, loss of normal feedwater, and loss of both normal and preferred power sources.

Chapter 4 of RESAR-SP/90 PDA Module 5, "Reactor System" discusses the design bases and design evaluation of core components. Details of the control and protection systems' instrumentation design and logic are discussed in Chapter 7 of RESAR-SP/90 PDA Module 9, "I&C and Electric Power". This information supports the accident analyses of Chapter 15 of RESAR-SP/90 PDA Modules 1, 4, 5, 6, 8, 10, 12, 13 & 16, which show that the acceptable fuel design limits are not exceeded for Condition 1 and 2 occurrences.

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."

Discussion

Whenever the reactor is critical, prompt compensatory reactivity feedback effects are assured by the negative fuel temperature effect (Doppler effect) and by the nonpositive operational limit on the moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is assured by the inherent design, using low enrichment fuel. The nonpositive moderator temperature coefficient of reactivity is assured by administratively controlling the dissolved absorber concentration or by using burnable absorbers.

Reactivity coefficients and their effects are discussed in Chapter 4 of RESAR-SP/90 PDA Module 5, "Reactor System".

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."

Discussion

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

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

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

If necessary to maintain axial imbalance within the limits of Chapter 16 of the integrated PDA document, (i.e., imbalances which are alarmed to the operator and are within the imbalance trip setpoints) the operator can suppress xenon axial oscillations by control rod motions and/or temporary power reductions.

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

The stability of the core against xenon-induced power oscillations and the functional requirements of instrumentation for monitoring and measuring core power distribution are discussed in Chapter 4 of RESAR-SP/90 PDA Module 5, "Reactor System". Details of the instrumentation design and logic are discussed in Chapter 7 of RESAR-SP/90 PDA Module 9, "I&C and Electrical Power".

Criterion 13 - Instrumentation and Control

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

Discussion

Instrumentation and controls are provided to monitor and control neutron flux, control rod position, fluid temperatures, pressures, flows, and levels, as necessary, to assure that adequate plant safety can be maintained. Instrumentation is provided in the reactor coolant system, steam and power conversion system, containment, engineered safety systems, radioactive waste management systems, and other auxiliary systems. Parameters that must be provided for operator use under normal operating and accident conditions are indicated in the control room in proximity to the controls for maintaining the indicated parameters in their proper ranges.

The quantity and types of process instrumentation provided ensure safe and orderly operation of all systems over the full design range of the plant. These systems are described in Chapters 6, 7, 8, 9, 10, 11 and 12 of RESAR-SP/90 PDA Modules 1, 6, 8, 9, 10, 11, 12 and 13.

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."

Discussion

The reactor coolant pressure boundary (RCPB) is designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation, including anticipated transients, with stresses within applicable limits. Consideration is given to loadings under normal operating conditions and to abnormal loadings, such as pipe rupture and seismic loadings, as discussed in Chapter 3. The piping is protected from overpressure by means of pressure-relieving devices, as required by American Society of Mechanical Engineers (ASME), Section III.

Reactor coolant pressure boundary materials and fabrication techniques are such that there is a low probability of gross rupture or significant leakage. (Refer to Criterion 31 for further discussion of reactor coolant pressure boundary.)

Coolant chemistry is controlled to protect the materials of construction of the RCPB from corrosion.

The RCPB welds are accessible for inservice inspections (ISI) to assess the structural and leaktight integrity. The details of the ISI program are given in the integrated RESAR-SP/90 PDA document. For the reactor vessel, a material surveillance program conforming to applicable codes is provided. Chapter 5 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System" has additional details.

Instrumentation is provided to detect significant leakage from the RCPB with indication in the control room, as discussed in Chapter 5 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System".

Criterion 15 - Reactor Coolant System Design

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

Discussion

Steady-state and transient analyses are performed to ensure that reactor coolant system (RCS) design conditions are not exceeded during normal operation. Protection and control setpoints are based on these analyses.

Additionally, RCPB components have a large margin of safety through application of proven materials and design codes, use of proven fabrication techniques, nondestructive shop testing, and integrated hydrostatic testing of assembled components.

The effect of radiation embrittlement is considered in reactor vessel design. Surveillance samples monitor adherence to expected conditions throughout the plant life.

Multiple safety and relief valves are provided for the RCS. These valves and their setpoints meet the ASME criteria for overpressure protection. The ASME criteria are satisfactory, based on a long history of industrial use. Chapter 5 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System" discusses the RCS design.

Criterion 16 - Containment Design

"The 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."

Discussion

A spherical steel containment structure encloses the entire RCS. It is designed to sustain, without loss of required integrity, the effects of LOCAs up to and including the double-ended rupture of the largest pipe in the RCS or double-ended rupture of a steam or feedwater pipe. Engineered safety features comprising the emergency core cooling system, containment spray system, and the containment air coolers serve to cool the reactor core and return the containment to near atmospheric pressure. The containment structure and engineered safety systems are designed to assure the required functional capability of containing any uncontrolled release of radioactivity. The radiological shielding and the containment limit the uncontrolled release of radioactivity to the environment.

Refer to RESAR-SP/90 PDA Modules 1, 4, 5, 6, 7, 8, 10, 12, 13, and 16.

Criterion 17 - Electrical Power Systems

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

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

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

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

Discussion

An onsite electric power system and an offsite electric power system are provided to permit the functioning of structures, systems, and components important to safety. As discussed in Chapter 8 of RESAR-SP/90 PDA Module 9 "I&C and Electrical Power", each Class 1E electric power system is designed with adequate independence, capacity, redundancy, and testability to ensure the functioning of engineered safety features (ESF). Independence is provided by physical separation and electrical isolation of components and cables.

The onsite AC power system includes a Class 1E system and a non-Class 1E system. Onsite AC power is supplied from the 230 kV switchyard through reserve auxiliary transformers which feed the non-Class 1E and Class 1E buses. The Class 1E AC power system is the power source used in (or associated with) shutting down the reactor and preventing or limiting the release of radioactive material following a design basis event. The system is divided into two independent ac power trains, train A and train B, each fed from an independent Class 1E bus.

Each Class 1E bus is provided with two (normal and alternate) offsite preferred power sources and a standby onsite power source. With both offsite sources available, each Class 1E bus is supplied from a separate reserve auxiliary transformer.

The Class 1E AC system distributes power to all safety related loads. Also, the Class 1E AC system supplies power to certain selected loads which are not safety related but are important to the plant operation; however, these loads are tripped when a safety injection signal is received.

The non-Class 1E AC system supplies preferred (offsite) power to the Class 1E AC system through the reserve auxiliary transformer 4160 V windings. Each reserve auxiliary transformer has the capacity to supply all connected non-Class 1E running loads and to start and run the loads of one Class 1E train. The offsite power systems are not included in the NPB.

A failure of a single component will not prevent the safety related systems from performing their function. Each of the preferred circuits is designed to be available in sufficient time, following a loss of all onsite power sources 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.

Emergency onsite AC power is furnished by two diesel-generators. Each diesel generator is connected to a Class 1E bus. The engineered safety feature (ESF) loads are divided between the Class 1E buses in balanced, redundant load groupings. Each diesel-generator is capable of supplying sufficient power in sufficient time for the operation of the ESF required for the unit; both diesel-generators start automatically. If preferred power is available to the Class 1E bus, the ESF loads will be started sequentially. However, in the event that preferred power is lost, the load sequencing system will shed all loads, connect each diesel-generator to its associated Class 1E bus, and sequentially start the ESF equipment. The diesel-generators are arranged so that a failure of a single component will not prevent the safe shutdown of the reactor. The onsite Class 1E DC power supply consists of four independent battery systems. Failure of a single component in the DC power supply will not impair function of the ESF required to maintain the reactor in a safe condition.

Criterion 18 - Inspection and Testing of Electric Power Systems

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

Discussion

Class 1E electric power systems are designed as described below in order that the following aspects of the system can be periodically tested:

A. The operability and functional performance of the components of Class 1E electric power systems (diesel generators, ESF buses, DC system).

B. The operability of these electric power systems as a whole and under conditions as close to design as practical, including the full operational sequence that actuates these systems.

The dc system is provided with detectors to indicate and alarm when there is a ground existing on any part of the system. During plant operation, normal maintenance may be performed.

Provisions for the testing of Class 1E AC electric power systems, Class 1E DC power systems, and the standby power supplies (diesel-generators) are described in Chapter 8 of RESAR-SP/90 PDA Module 9, "I&C and Electrical Power". Inspection and testing of the offsite power systems are not included in the NPB.

Criterion 19 - Control Room

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

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

Discussion

A control room is provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe manner under accident conditions. Operator action outside of the control room to mitigate the consequences of an accident is permitted. The control room and its post-accident ventilation systems are designed to satisfy Seismic Category 1 requirements, as discussed in Chapter 3. Adequate shielding and radiation protection are provided against direct gamma radiation and inhalation doses resulting from a postulated release of fission products inside the containment structure based on the assumptions contained in Regulatory Guide 1.4. The shielding and the control room standby air-conditioning system allow access to and occupancy of the control rooms under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. (Refer to Chapter 15 of RESAR-SP/90 PDA Modules 1, 4, 5, 6, 8, 10, 12, 13 and 16.) Fission product removal is provided in the control room recirculation equipment to remove iodine and particulate matter, thereby minimizing the control thyroid dose which could result from the accident. The control room habitability features are described in Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10.

In the event that the operators are forced to abandon the control room, panel-mounted instrumentation and controls are provided on the train-related shutdown panels to achieve and maintain the plant in the safe shutdown condition. (See Section 7.4 of RESAR-SP/90 PDA Module 9, "I&C and Electrical Power").

3.1.3 Protection and Reactivity Control Systems

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."

Discussion

A fully automatic protection system with appropriate redundant channels is provided to cope with transient events where insufficient time is available for manual corrective action. The design basis for all protection systems is in accordance with the guidelines of Institute of Electrical and Electronic Engineers (IEEE) Standards 279-1971 and 379-1972. The reactor protection system automatically initiates a reactor trip when any variable monitored by the system or combination of monitored variables exceeds the normal operating range. Setpoints are designed to provide an envelope of safe operating conditions with adequate margin for uncertainties to ensure that the fuel design limits are not exceeded.

Reactor trip is initiated by removing power to the rod drive mechanisms of all the rod cluster control assemblies. This causes the rods to insert by gravity, thus rapidly reducing the reactor power. The response and adequacy of the protection system have been verified by analysis of anticipated transients.

The ESF actuation system automatically initiates emergency core cooling and other safety functions by sensing accident conditions, using redundant analog channels measuring diverse variables. Manual actuation of safety features may be performed where ample time is available for operator action. The ESF actuation system automatically trips the reactor on a manual or automatic safety injection signal.

Criterion 21 - Protection System Reliability and Testability

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

Discussion

The protection system is designed for functional reliability and inservice testability. The design employs redundant logic trains and measurement and equipment diversity.

The protection system, including the ESF test cabinet, is designed to meet Regulatory Guide 1.22 and conform to the requirements of IEEE Standards 279-1971 and 379-1972. Functions that cannot be tested with the reactor at power are tested during shutdown, as allowed by the regulatory guide and these standards.

In cases where actuated equipment cannot be tested at power, the channels and logic associated with this equipment, up to the final actuation device, have the capability for testing at power. Such testing discloses failures or reductions in redundancy which may have occurred.

Removal from service of any single channel or component does not result in the loss of minimum required redundancy. For example, a two-of-three function is placed in the one-of-two mode when one channel is removed. (Note that distinction is made between channels and trains in this discussion. A train may be removed from service only during testing.) Bypassed and inoperable status indication for safety-related systems is provided in accordance with Regulatory Guide 1.47

Semiautomatic testers are built into each of the two logic trains of the protection system. These testers have the capability of testing the system logic very rapidly while the reactor is at power. A self-testing provision is designed into each tester. (For a detailed description of reliability and testability of the protection system, refer to Section 7.2 of RESAR-SP/90 PDA Module 9, "I&C Electrical Power".

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 the 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."

Discussion

Design of the protection systems includes consideration of natural phenomena, normal maintenance, testing, and accident conditions so that the protection functions are always available.

Protection system components are designed, arranged, and qualified for operation in the environment accompanying any emergency situation in which the components are required to function.

Functional diversity has been designed into the system. The extent of this functional diversity has been evaluated for a variety of postulated accidents. Diverse protection functions will automatically terminate an accident before intolerable consequences can occur.

Sufficient redundancy and independence are designed into the protection systems to assure that no single failure or removal from service of any component or channel of a system would result in loss of the protection function. Functional diversity and consequential location diversity are designed into the system. Automatic reactor trips are based upon neutron flux measurements, reactor coolant loop temperature measurements, pressurizer pressure and level measurements, and reactor coolant pump power supply underfrequency, undervoltage measurements, and other parameters. Trips may also be initiated manually or by a safety injection signal. See RESAR-SP/90 PDA Module 9, "I&C Electrical Power" for details.

High quality components, conservative design and applicable quality control, inspection, calibration, and tests are utilized to guard against common-mode failure. Qualification testing and analysis is performed on the various safety systems to demonstrate functional operation at normal and post-accident conditions of temperature, humidity, pressure, and radiation for specified periods, if required. Typical protection system equipment is subjected to type tests under simulated seismic conditions, using conservatively large accelerations and applicable frequencies. The test results indicate no loss of the protection function. (Refer to Sections 3.10 and 3.11 for further details).

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."

Discussion

The protection system is designed with consideration of the most probable failure modes of the components under various perturbations of the environment and energy sources. Each reactor trip channel is designed on the deenergize-to-trip principle so loss of power, disconnection, open channel faults, and the majority of the internal channel short circuit faults cause the channel to go into its tripped mode.

Similarly, that portion of the ESF actuation system provided for actuation of the emergency feedwater system and containment ventilation isolation is designed to fail into a safe state, except for the final output relays. The relays are energized to actuate, as are the pumps and motor-operated valves of the actuated equipment.

For a more detailed description of the protection system, refer to Chapter 7 of RESAR-SP/90-PDA Module 9, "I&C Electrical Power".

Criterion 24 - Separation of Protection and Control Systems

"The protection system shall be separated from the 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."

Discussion

The protection system is separate and distinct from the control systems, as described in Chapter 7 of RESAR-SP/90 PDA Module 9, "I&C and Electrical Power". Control systems are, in some cases, dependent on the protection system in that control signals are derived from protection system measurements, where applicable. These signals are transferred to the control system by isolation devices which are classified as protection components. The adequacy of the system isolation has been verified by testing under conditions of postulated credible faults. The failure of any single control system component or channel, or the failure or removal from service of any single protection system component or channel which is common to the control and protection system, leaves intact a system which satisfies the requirements of the protection system. The removal of a train from service is allowed only during testing of the train. Distinction between channel and train is made in the discussions.

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 the control rods."

Discussion

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

Analyses of the effects of possible malfunctions are discussed in Chapter 15 of RESAR-SP PDA Modules 1, 4, 5, 6, 8, 10, 12, 13 and 16. These analyses show that for postulated boron dilution during refueling, start-up, or manual or automatic operation at power, the operator has ample time to determine the cause of dilution, terminate the source of dilution, and initiate reboration before the shutdown margin is lost. The analyses show that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

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 that the acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions."

Discussion

Two reactivity control systems are provided. These are rod cluster control assemblies and gray rod assemblies (RCCAs and GRAs) and chemical shim (boric acid). The RCCAs/GRAs are inserted into the core by the force of gravity.

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

The boron system will maintain the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

Details of the construction of the RCCAs/GRAs and their operation are discussed in Chapters 3 and 4 of RESAR-SP/90 PDA Module 5, "Reactor Systems". The means of controlling the boric acid concentration is described in Chapter 9 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems". Performance analyses under accident conditions are included in Chapter 15 of RESAR-SP/90 PDA Modules 1, 4, 5, 6, 8, 10, 12, 13 and 16.

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."

Discussion

The facility is provided with means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. These means are discussed in detail in Chapters 4 and 9 of RESAR-SP/90 PDA Module 5, "Reactor Systems" and Module 13, "Auxiliary Systems", respectively. Combined use of the Control Rods/GRAs and the chemical shim control system permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. The single highest worth Control Rod/GRA is assumed to be stuck full out upon trip for this determination.

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."

Discussion

The maximum reactivity worth of the control rods and the maximum rates of reactivity insertion employing control rods and boron removal are limited to values that prevent any reactivity increase from rupturing the RCS boundary or disrupting the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

The appropriate reactivity insertion rate for the withdrawal of control rods and the dilution of the boric acid in the reactor coolant systems are specified in the Technical Specifications. The specification includes appropriate graphs that show the permissible withdrawal limits and overlap of the control rod banks as a function of power. These data on reactivity insertion rates, dilution, and withdrawal limits are also discussed in Chapter 4 of RESAR-SP/90 PDA Module 5, "Reactor Systems". The capability of the chemical and volume control system to avoid an inadvertent excessive rate of boron dilution is discussed in Chapter 9 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems". The relationship of the reactivity insertion rates to plant safety is discussed in Chapter 15 of RESAR-SP/90 PDA Modules 1, 4, 5, 6, 8, 10, 12, 13, and 16.

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

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."

Discussion

The protection and reactivity control systems have an extremely high probability of performing their required safety functions in any anticipated operational occurrences. Diversity and redundancy, coupled with a quality assurance program and analyses, support this probability as does operating experience in plants using the same basic design. Failure modes of system components are designed to be safe modes. Loss of power to the protection system results in a reactor trip. Details of system design are covered in Chapters 4 and 7 of RESAR-SP/90 PDA Module 5, "Reactor Systems" and Module 9, "Auxiliary Systems", respectively.

3.1.4 Fluid Systems

Criterion 30 - Quality of Reactor Coolant Pressure Boundary

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

Discussion

All RCS components are designed, fabricated, inspected, and tested in conformance with the ASME Boiler and Pressure Vessel Code, Section III.

All balance of plant components are classified according to Regulatory Guide 1.26, and all nuclear steam supply system (NSSS) components are classified according to ANS-51.1, 1983 (which is an acceptable alternative to Regulatory Guide 1.26) and are accorded all the quality measures appropriate to these classifications. The design bases and evaluations of the RCS are discussed in Chapter 5 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System".

A number of methods are available for detecting reactor coolant leakage. The reactor vessel closure joint is provided with a temperature monitored leakoff between double gaskets. Leakage inside the reactor containment is drained to the containment building and reactor cavity sumps, where the level is monitored. Leakage is also detected by measuring the airborne activity and humidity of the containment. Monitoring the inventory of reactor coolant in the system at the pressurizer, volume control tank, and reactor coolant drain tank provides an accurate indication of integrated leakage. Refer to Chapter 5 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System" for complete description of the RCPB leakage detection system.

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."

Discussion

Close control is maintained over material selection and fabrication for the RCS to assure that the boundary behaves in a nonbrittle manner. The RCS materials which are exposed to the coolant are corrosion-resistant stainless steel or Inconel. The nil ductility transition reference temperature (RT_{NDT}) of the reactor vessel structural steel is established by Charpy V-notch and drop weight tests in accordance with 10CFR50, Appendix G, "Fracture Toughness Requirements".

The reactor vessel specification imposes the following requirements which are not specified by the ASME code.

A. The performance of a 100-percent volumetric ultrasonic shear wave test of reactor vessel plate and a post-hydrotest ultrasonic map of all welds in the pressure vessel are required. Cladding bond ultrasonic inspection to more restrictive requirements than those specified in the code is also required to preclude interpretation problems during inservice inspection.

B. In the surveillance programs, the evaluation of the radiation damage is based on preirradiation testing of Charpy V-notch and tensile specimens and postirradiation testing of Charpy V-notch, tensile, and 1/2 T compact tension specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the reference transition temperature approach and the fracture mechanics approach, and are in accordance with American Society of Testing Materials (ASTM) E-185, Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, and the requirements of 10CFR50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements".

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

The fabrication and quality control techniques used in the fabrication of the RCS are equivalent to those used for the reactor vessel. The inspections of reactor vessel, pressurizer, piping, pumps, and steam generators are governed by ASME code requirements. (Refer to Chapter 5 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System" for details).

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

Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

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

Discussion

The design of the RCPB provides accessibility to the entire internal surfaces of the reactor vessel and most external zones of the vessel, including the nozzle to reactor coolant piping welds, the top and bottom heads, and external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete. The inspection capability complements the leakage detection systems in assessing the pressure boundary components' integrity. The RCPB will be periodically inspected under the provisions of the ASME Code, Section XI.

Monitoring of changes in the fracture toughness properties of the reactor vessel core region plates, forgings, weldments, and associated heat-treated zones is performed in accordance with 10CFR50, Appendix H. Samples of reactor vessel plate materials are retained and catalogued in case future engineering development shows the need for further testing.

The material properties surveillance program includes conventional tensile and impact tests, and fracture mechanics specimens. The observed shifts in RT_{NDT} of the core region materials with irradiation will be used to confirm the allowable limits calculated for all operational transients.

The design of the RCPB piping provides for accessibility of all welds requiring inservice inspection under the provisions of the ASME Code, Section XI. Removable insulation is provided at all welds requiring inservice inspection.

Criterion 33 - Reactor Coolant Makeup

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

Discussion

The chemical and volume control system provides a means of reactor coolant make up and adjustment of the boric acid concentration. Make up is added automatically if the level in the volume control tank falls below a preset level. The positive displacement charging pump is used as the normal means of reactor coolant make up. This pump is powered from the offsite power system.

The centrifugal charging pumps are a backup method of providing reactor coolant make up. The centrifugal charging pumps are capable of supplying the required make up and reactor coolant seal injection flow when power is available from either onsite or offsite electric power systems. Functional reliability is assured by provision of standby components assuring a safe response to probable modes of failure. Details of system design, including descriptions of the effects of small piping and component ruptures, are provided in Sections 6.3 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System" and Section 9.3 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems" and in Chapter 15 of RESAR-SP/90 PDA Modules 1, 4, 5, 6, 8, 10, 12, 13, and 16. Details of the electric power system are included in Chapter 8 of RESAR-SP/90 PDA Module 9, "I&C Electrical Power".

Criterion 34 - Residual Heat Removal

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

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

Discussion

The residual heat removal (RHR) portion of the WAPWR integrated safeguards system, in conjunction with the steam and power conversion system, is designed to transfer the fission product decay heat and other residual heat from the reactor core at a rate which keeps the fuel within acceptable limits. The RHR system functions when temperature and pressure are below approximately 350°F and 400 psig, respectively.

Redundancy of the RHR system is provided by two residual heat removal pumps (located in separate flood-proof compartments, with means available for draining and monitoring leakage), two heat exchangers, and associated piping, cabling, and electric power sources. (For a more detailed description of RHR system redundancy, refer to Subsection 5.4.7 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System".) The RHR system is able to operate on either the onsite or offsite electrical power system.

Redundancy of heat removal at temperatures above approximately 350°F is provided by the four steam generators, atmospheric relief valves, and the emergency feedwater system.

Details of the system design are provided in Subsection 5.4.7, Chapter 9, and Chapter 10 of RESAR-SP/90 PDA Modules 1, 13 and, 6 and 8, respectively.

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 electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished. assuming a single failure."

Discussion

The emergency core cooling portion of the ISS has the capability to mitigate the effects of any LOCA within the design bases. Cooling water is provided in an emergency to transfer heat from the core at a rate sufficient to maintain the core in a coolable geometry and to assure that clad metal-water reaction is limited to less than 1 percent. Design provisions assure performance of the required safety functions even with a postulated single failure.

Emergency core cooling is provided even if there should be a failure of any component in the system. A passive system of three accumulators which do not require any external signals or source of power to operate provide the short term cooling requirements for reactor coolant pipe system breaks. Three independent and redundant pumping systems are provided: the charging system, safety injection system, and residual heat removal system. The charging system is a high pressure, low flow system capable of providing the required emergency cooling for small breaks. The safety injection system is an intermediate pressure, intermediate flow system capable of providing the required emergency cooling for medium-sized breaks. The charging system can be operated to complement the safety injection system. The RHR system is a low pressure, high flow system capable of providing the required emergency cooling for large breaks. The charging system and safety injection system can be operated to complement the RHR system. These systems are arranged so that the single failure of any active component does not interfere with meeting the short term cooling requirements.

The primary function of the ECCS is to deliver borated cooling water to the reactor core in the event of a LOCA. This limits the fuel-clad temperature, ensures that the core will remain intact and in place with its essential heat transfer geometry preserved, and prevents a return to criticality. This protection is afforded for:

A. All pipe breaks sizes up to and including the hypothetical circumferential rupture of the largest pipe of a reactor coolant loop.

B. A loss of coolant associated with a rod ejection accident.

The ECCS portion of the WAPWR ISS is described in Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10. The LOCA, including an evaluation of consequences, is discussed in Chapter 15 of RESAR-SP/90 PDA Modules 1, 4, 5, 6, 8, 10, 12, 13, and 16.

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."

Discussion

The ECCS is accessible for visual inspection and for nondestructive inservice inspection to satisfy the ASME Code, Section XI. Components outside the containment are accessible for leaktightness inspection during operation of the reactor.

Criterion 37 - Testing of Emergency Core Cooling System

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

Discussion

The design of the ECCS permits periodic testing of both active and passive components of the ECCS. Preoperational performance tests of the ECCS components are performed by the manufacturer. Initial system hydrostatic and functional flow tests demonstrate structural and leaktight integrity of components and proper functioning of the system. Thereafter, periodic tests demonstrate that components are functioning properly.

Each active component of the ECCS can be individually operated on the normal power source or transferred to standby power sources at any time during normal plant operation to demonstrate operability. The centrifugal charging pumps are not normally operating but, as part of the charging system, they are available for operation as necessary during plant operation. The test of the safety injection pumps employs the minimum flow recirculation test line which connects back to the refueling water storage tank. Remote-operated valves are exercised and actuation circuits tested. The automatic actuation circuitry, valves, and pump breakers can be checked during integrated system tests performed during a planned cooldown of the RCS.

Design provisions include special instrumentation, testing, and sampling lines to perform the tests during plant shutdown to demonstrate proper automatic operation of the ECCS. A test signal is applied to initiate automatic action, and verification is made that the safety injection pumps attain required discharge heads. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry. In addition, the periodic recirculation to the refueling water storage tank verifies the ECCS delivery capability. This recirculation test includes all but the last valve, which connects to the reactor coolant piping.

The design provides for capability to test initially, to the extent practical, the full operational sequence up to the design conditions, including transfer to alternate power sources for the ECCS to demonstrate the state of readiness and capability of the system. This functional test is performed with the water level below the safety injection signal setpoint in the pressurizer and with the RCS initially cold and depressurized. The ECCS valving is set initially to simulate the system alignment for plant power operation.

Details of the ECCS are found in Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10. Performance under accident conditions is evaluated in Chapter 15 of RESAR-SP/90 PDA Modules 1, 4, 5, 6, 8, 10, 12, 13, and 16. Surveillance requirements are identified in the Technical Specifications.

Criterion 38 - Containment Heat Removal System

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

"Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite 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."

Discussion

The containment spray and containment fan cooler systems, in conjunction with the ECCS, are capable of removing sufficient energy and subsequent decay energy from the containment following the hypothesized LOCA to maintain the containment pressure below the containment design pressure. During the post-accident injection phase, water for the containment spray system and ECCS is drawn from the refueling water storage tank. During the later recirculation phase, spray water and ECCS water are pumped from the containment sumps.

Each of the containment spray and containment fan cooler systems consists of two independent subsystems supplied from separate Class 1E power buses. No single failure, including loss of onsite or offsite electrical power, can cause loss of more than half of the installed 200-percent cooling capacity. The containment spray system and containment fan coolers are discussed in Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10. Electrical facilities are described in Chapter 8 of RESAR-SP/90 PDA Module 9, "I&C Electrical Power". A containment pressure and temperature analysis following a LOCA is given in Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10, with additional results found in Chapter 15 of RESAR-SP/90 PDA Modules 1, 4, 5, 6, 8, 10, 12, 13, and 16.

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."

Discussion

The essential equipment of the containment spray system (CSS) is outside the containment, except for risers, distribution header piping, spray nozzles, and the containment sumps. The containment sumps, spray piping, and nozzles can be inspected during shutdown. Portions of the containment spray suction piping and the RHR suction piping from the containment recirculation sumps are not accessible for inspection. Associated equipment outside the containment can be visually inspected.

The containment air coolers and associated cooling water system piping inside the containment can be inspected during shutdowns.

These periodic inspections assure that the capability of these heat removal systems as specified in the Technical Specifications is met.

For details on the containment air coolers and containment spray system, see Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10.

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 "

Discussion

The containment spray system and the containment fan cooling system are designed to permit periodic testing to assure the structural and leaktight integrity of their components and to assure the operability and performance of the active components of the systems. All active components of the CSS and delivery piping up to the last powered valve before the spray nozzle have the capability to be tested during reactor power operation. In addition, when the unit is shut down, smoke or air can be blown through the test connections for visual verification of the flow path. All safety related active components of the containment fan cooling system can be tested to verify operability during reactor power operation. In addition, since the containment fan cooling system is a normally operating system, the performance and operability of portions of the system are continuously verified during normal reactor power operation. The facility design allows, under conditions as close to the design as practicable, the performance of a full operational sequence that brings these systems into operation. More complete discussions of the testing of these systems are in Chapters 6 of RESAR-SP/90 PDA Modules 1 and 10, and the Technical Specifications.

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."

Discussion

The CSS serves to remove radioiodine and other airborne particulate fission products from the containment atmosphere following a postulated LOCA. The system consists of two independent systems, each supplied from separate electrical power buses, as described in Chapter 8 of RESAR-SP/90 PDA Module 9, "I&C and Electrical Power". Either subsystem alone can provide the fission product removal capacity for which credit is taken in Chapter 15, of RESAR-SP/90 PDA Modules 1, 4, 5, 6, 8, 10, 12, 13, and 16, in conformance with Regulatory Guide 1.4.

The generation of hydrogen in the containment under post-accident conditions has been evaluated, using the assumptions of Regulatory Guide 1.7. (See Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10.) A post-accident hydrogen recombiner system is provided with redundancy of vital components so that a single failure does not prevent timely operation of the system. This system is described in Subsection 6.2.5 of RESAR-SP/90 PDA Module 10, "Containment Systems". The post-LOCA purge exhaust system is provided as a backup. No single failure causes both subsystems to fail to operate.

Criterion 42 - Inspection of Containment Atmosphere Cleanup System

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

Discussion

The containment atmosphere cleanup systems are designed and located so that they can be inspected periodically, as required. The essential equipment of the CSS is outside the containment, except for risers, distribution header piping, and spray nozzles in the containment. The hydrogen recombiners are located inside the containment. The post LOCA purge exhaust filter unit and the hydrogen monitors are located outside the containment. The equipment outside the containment may be inspected during normal power operation. Components of the CSS, the post LOCA purge exhaust system, and the hydrogen recombiner and monitoring system located inside the containment, can be inspected during refueling shutdowns. (See Chapter 6 of RESAR/SP-90 PDA Modules 1 and 10 for details on these systems.)

Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

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

Discussion

The CSS which serves as the containment atmosphere cleanup system can be tested. The operation of the spray pumps can be tested by recirculation to the refueling water storage tank through a test line. The system valves can be operated through their full travel. The system is checked for leaktightness during testing. See Subsection 6.2.2.2 of RESAR-SP/90 PDA Modules 1 and 10, for details and Chapter 8 of RESAR-SP/90 PDA Module 9, "I&C Electrical Power" for electrical power details. The spray headers and nozzles can be smoke or air tested, as described in the response to Criterion 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 redundance in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure."

Discussion

The component cooling water (CCW) and service water (SW) systems are provided to transfer heat from plant safety-related components to the ultimate heat sink. These systems are designed to transfer their respective heat loads under all anticipated normal and accident conditions. Suitable redundancy, leak detection, systems interconnection, and isolation capabilities are incorporated in the design of these systems to assure the required safety function, assuming a single failure, with either onsite or offsite power.

A complete description of the CCW system is given in Chapter 9 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems".

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."

Discussion

The CCW system is capable of being monitored during normal operation. 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. The integrity of any underground piping will be demonstrated by pressure and functional tests. Piping to and from the containment air coolers is accessible for inspection during reactor shutdown and refueling periods. This system is discussed in Chapter 9 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems". Inspection of the SW system is not part of the NPB. The SW system is not part of the Nuclear Power Block scope of supply.

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 LOCA, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources."

Discussion

The CCW system operates continuously during normal plant operation and shutdown under flow and pressure conditions that approximate the accident conditions. These operations demonstrate the operability, performance, and structural and leaktight integrity of all cooling water system components.

The cooling water system is designed to include the capability for testing through the full operational sequence that brings the system into operation for reactor shutdown and for LOCAs, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources. The CCW system is capable of being tested during normal operation by alternating operation of the systems between the redundant trains.

For a detailed description of the cooling water systems, refer to Section 9.2 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems". Testing of the SW system is not part of the NPB.

3.1.5 Reactor Containment

Criterion 50 - Containment Design Basis

"The reactor containment structure, including access opening, 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 degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters."

Discussion

The design of the containment structure is based on the containment design basis accidents which include the rupture of a reactor coolant pipe in the RCS or the rupture of a main steam line. In either case, the pipe rupture is assumed to be coupled with partial loss of the redundant safety feature systems (minimum safety features). The maximum pressure and temperature reached for a containment design basis accident are presented in Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10. The containment design, as discussed in Subsection 3.8.2, provides ample margin to the design basis limits.

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."

Discussion

Principal load-carrying components of ferritic materials exposed to the external environment are selected (as discussed in Subsection 3.8.2) so that their temperatures under normal operating and testing conditions are not less than 30°F above nil ductility transition temperature.

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."

Discussion

The containment system is designed and constructed and the necessary equipment is provided to permit periodic integrated leakage rate tests during plant lifetime in accordance with the requirements of Appendix J of 10CFR50. Details concerning the conduct of periodic integrated leakage rate tests are included in Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10.

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."

Discussion

Provisions exist for conducting individual leakage rate tests on containment penetrations. Penetrations are visually inspected and pressure tested for leaktightness at periodic intervals. Other inspections are performed as required by Appendix J of 10CFR50. (Refer to Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10.)

Criterion 54 - Piping Systems Penetrating Containment

"Piping systems penetrating the 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."

Discussion

Piping systems penetrating the primary reactor containment are provided with containment isolation valves. Penetrations which must be closed for containment isolation have redundant valving and associated apparatus. Automatic isolation valves with air or motor operators, which do not restrict normal plant operation, are periodically tested to assure operability. Secondary system piping inside the containment is considered an extension of the containment boundary, as described in Subsection 6.2.4 of RESAR-SP/90 PDA Module 10, "Containment Systems". The isolation valve arrangements are discussed in Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10.

Piping that penetrates the containment has been equipped with test connections and test vents or has other provisions to allow periodic leak rate testing to ensure that leakage is within the acceptable limit as defined by the Technical Specifications and Appendix J to 10CFR50, as described in Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10.

The fuel transfer tube is not classified as a fluid system penetration. The blind flange and the portion of the transfer tube inside the containment are an extension of the containment boundary. The blind flange isolates the transfer tube at all times, except when the reactor is shutdown for refueling. This assembly is a penetration in the same sense as are equipment hatches and personnel locks.

Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

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

1. One locked closed isolation valve inside and one locked closed isolation valve outside the containment; or

2. One automatic isolation valve inside and one locked closed isolation valve outside the containment; or

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

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

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

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

Discussion

Each line that is a part of the reactor coolant pressure boundary and penetrates the containment is provided with isolation valves meeting the intent of this criterion, except that the reactor shutdown cooling lines (RHR system) which are part of the RCPB and which penetrate the containment are provided with two isolation valves in series, both inside the containment. This system is a closed system outside the containment and is constructed to ASME Code, Section III, Class 2 specifications and is considered the second passive barrier to fission product release, as described in Chapter 6 of

RESAR-SP/90 PDA Modules 1 and 10. The arrangement and type of valves utilized are discussed in Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10. Containment penetrations are Seismic Category 1 and are protected against possible environmental effects, including missiles.

Criterion 56 - Primary Containment Isolation

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

1. One locked closed isolation valve inside and one locked closed isolation valve outside the containment; or

2. One automatic isolation valve inside and one locked closed isolation valve outside the containment; or

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

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

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

Discussion

Lines which communicate directly with the containment atmosphere and which penetrate the reactor containment are normally provided with two isolation valves in series, one inside and one outside the containment, in accordance with one of the above acceptable arrangements. Several penetrations use alternative arrangements which satisfy containment isolation on some other defined bases. Special cases are described in Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10.

Valving arrangements are combinations of locked-shut isolation valves and automatic isolation valves or remote-manual isolation valves. No simple check valves are utilized as automatic isolation valves outside the containment. Where necessary, provision for leak detection is provided for lines outside the containment.

Instrument lines satisfy other acceptable criteria, as described in Chapter 6 of RESAR-SP/90 PDA Modules 1 and 10.

Criterion 57 - Closed System Isolation Valves

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

Discussion

Lines which penetrate the containment and are neither part of the RCPB nor connected directly to the containment atmosphere are considered closed systems within the containment and are equipped with at least one containment isolation valve of one of the following types:

A. An automatic isolation valve (a simple check valve is not used as this automatic valve).

B. A locked-closed valve.

C. A valve capable of remote manual operation.

This valve is located outside the containment and as close to the containment wall as practical. Valve locations are discussed in detail in Subsection 6.2.4 of RESAR-SP/90 PDA Module 10.

3.1.6 Fuel and Reactivity Control

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

"The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for the 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."

Discussion

Means are provided to control 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. The radioactive waste management systems are designed to minimize the potential for an inadvertent release of radioactivity from the facility and to assure that the discharge of radioactive wastes is maintained as low as practicable below regulatory limits of 10CFR20 during normal operation. The radioactive waste management systems, the design criteria, and the amounts of estimated releases of radioactive effluents to the environment are described in Chapter 11 of RESAR-SP/90 Module 12, "Waste Management".

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."

Discussion

The spent fuel pool and associated cooling system, fuel handling system, and radioactive waste processing system are designed to assure adequate safety under normal and postulated accident conditions.

The spent fuel pool cooling system provides cooling to remove residual heat from the fuel stored in the spent fuel pool. The system is designed with redundancy and testability to assure continued heat removal. The spent fuel pool cooling system is described in Subsection 9.1.3 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems". The spent fuel pool is designed so that no postulated accident could cause excessive loss-of-coolant inventory. Accidents are discussed in Chapter 15 of RESAR-SP/90 PDA Modules 1, 4, 5, 6, 8, 10, 12, 13, and 16.

Structures, components, and systems are designed and located so that appropriate periodic inspection and testing may be performed.

Adequate shielding is provided as described in Chapter 12 of RESAR-SP/90 PDA Module 11, "Radiation Protection". Radiation monitoring is provided as discussed in Chapters 11 and 12 of RESAR-SP/90 PDA Module 12, "Waste Management" and Module 11, "Radiation Protection", respectively.

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

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."

Discussion

The restraints and interlocks provided for the safe handling and storage of new and spent fuel are discussed and illustrated in Chapter 9 of RESAR-SP/90 PDA Module 13, "Auxiliary Systems".

Criticality in new and spent fuel storage areas is prevented both by physical separation of fuel assemblies and the presence of borated water in the spent fuel storage pool. The center-to-center distance between the adjacent spent fuel assemblies is sufficient to ensure a $k_{eff} < 0.95$, even if unborated water is used to fill the spent fuel storage pool. New fuel is stored with enough center-to-center distance to ensure a $k_{eff} < 0.98$, under conditions of optimum moderation.

The design of the spent fuel storage rack assembly is such that it is configurationally impossible to insert the spent fuel assemblies in other than prescribed locations, without physically modifying the rack, thereby preventing any possibility of accidental criticality.

Layout of the fuel handling area is such that the spent fuel cask cannot traverse the spent fuel storage pool.

Criterion 63 - Monitoring Fuel and Waste Storage

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

Discussion

Instrumentation is provided to detect and alarm in the control room excessive temperature or low water level in the spent fuel storage pool. Area radiation monitors are provided in the fuel storage area for personnel protection and general surveillance. These area monitors alarm locally and in the control room. Normally, the fuel building ventilation system removes radioactivity from the atmosphere above the spent fuel storage pool and discharges it by way of the plant vent. The ventilation system is continuously monitored by gaseous, particulate, and radioiodine radiation monitors.

If radiation levels reach a predetermined point, an alarm is sounded in the control room and the ventilation discharge path is automatically transferred through filter adsorber units which provide adequate filtration before discharge from the plant vent. (See Chapters 7, 9, and 12 of RESAR-SP/90 PDA Modules 9, "I&C Electrical Power", 13, "Auxiliary Systems", and 11, "Radiation Protection", respectively, for details).

Criterion 64 - Monitoring Radioactivity Releases

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

Discussion

The containment atmosphere is continually monitored during normal and transient station operations using the containment particulate, gaseous, and radioiodine radiation monitors. Under accident conditions, samples of the containment atmosphere provide data on existing airborne radioactive concentrations within the containment. Area radiation monitors located in auxiliary buildings are provided for continual monitoring of radiation levels in the spaces which contain components for recirculation of LOCA fluids and components for processing radioactive wastes. Radioactivity levels contained in the facility effluent and discharge paths and in the plant environs are continually monitored during normal and accident conditions by the plant radiation monitoring systems. In addition to the installed detectors, periodic plant environmental surveillance is established. Measurement capability and reporting of effluents are based on the guidelines of Regulatory Guides 1.4 and 1.21. Radiation monitoring systems are discussed in Section 11.5 and Subsection 12.3.4 of RESAR-SP/90 PDA Module 12, "Waste Management" and Module 11, "Radiation Protection", respectively.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

This section provides a guide to the classification method of structures, components, and systems.

3.2.1 Seismic Classification

General Design Criterion 2 of Appendix A to 10CFR50, General Design Criteria for Nuclear Power Plants, requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform necessary safety functions. Appendix A to 10CFR100, Seismic and Geologic Siting Criteria for Nuclear Power Plants, sets forth the principal seismic and geologic considerations which are used in the evaluation of the suitability of plant design bases established in consideration of the site seismic and geologic characteristics.

3.2.1.1 Definitions

Seismic Category I structures, components, and systems are classified in accordance with Regulatory Guide 1.29. Safety-related, Seismic Category I structures, components, and systems are those necessary to ensure the following:

A. The integrity of the reactor coolant pressure boundary.

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

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

Seismic Category I structures, components, and systems are designed to withstand the appropriate seismic loads, as discussed in Sections 3.7 through 3.11 of this module and other applicable loads without loss of function. Seismic Category I structures are sufficiently isolated from non-Category I structures, or they are analyzed to ensure that their structural integrity is maintained during the postulated safe shutdown earthquake (SSE). Non-Seismic Category I systems, equipment, and components installed in Seismic Category I structures whose failure could result in loss of required safety function of Seismic Category I structures, equipment, systems, or components are either separated by distance or barrier from the affected structure, system, equipment, or component or designed together with their anchorages to maintain their structural integrity during the SSE.

Structures, equipment, and systems not classified as Seismic Category I are classified as Seismic Category II.

Safety-related structures, systems, and components that are classified Seismic Category I are in compliance with the quality assurance requirements of 10CFR50, Appendix B.

The criteria used for the NPB design of (a) Seismic Category I structures, equipment, systems, and components, (b) Seismic Category II items, and (c) Seismic Category II items whose failure could result in the loss of required safety function of Seismic Category I items are discussed in Section 3.7 of this module.

3.2.1.2 Classifications

Table 3.2-1 provides a listing of structures, components, and systems within the NPB and identifies those that are Seismic Category I.

Where only portions of systems are identified as Seismic Category I in Table 3.2-1, the boundaries of the Seismic Category I portions of the system will be shown on the piping and instrumentation diagrams in appropriate sections of the RESAR-SP/90 FDA version.

3.2.2 Classification System

Equipment, components, and structures in the WAPWR NPB are categorized according to nuclear safety, seismic category, codes, and standards. This system conforms to 10CFR50, ANS-51.1-1983, and Regulatory Guides 1.26 and 1.29. A three element classification is assigned to each item and indicates, in sequence, the nuclear safety class, the seismic category, and the applicable codes and standards. The classification system provides an easily recognizable means of identifying the extent to which the NPB components, equipment, and structures are related to nuclear safety and seismic qualification requirements. In addition, the classification system provides the means whereby the codes and/or standards that govern the design of a component or structure can be located. Table 3.2-1 provides a listing of the principal structures, systems, components, and their associated classifications.

3.2.2.1 Nuclear Safety Classifications

The first element of the classification identifies the nuclear safety class. The nuclear safety classifications designators used on the WAPWR NPB are as defined in ANSI/ANS-51.1-1983 for SC-1, SC-2, SC-3, and NNS.

3.2.2.2 Seismic Classification

The second element of the classification is either I or II, which designates the appropriate seismic category. Seismic classification is discussed in Subsection 3.2.1 of this module.

3.2.2.3 Codes and Standards

The third element of the classification indicates the principal codes and/or standards applicable to plant equipment, components, and structures. The codes and standards designators used on the WAPWR NPB are as follows:

Designator	Codes and Standards
1	American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code, Section III, Class 1.
2	ASME B&PV Code, Section III, Class 2.
3	ASME B&PV Code, Section III, Class 3.
4	Regulatory Guide 1.26 - Table 1 - Quality Group D.
5	Regulatory Guide 1.143 - Table 1. The codes and standards used for the construction of radioactive waste management and steam generator blowdown systems are provided in Regulatory Guide 1.143. Quality assurance requirements are to be applied to radioactive waste management systems as described in Regulatory Guide 1.143.
6	ASME B&PV Code, Section I.
7	Applicable National Fire Protection Association (NFPA) codes as qualified by Branch Technical Position APCSB 9.5-1, Appendix A. The design, fabrication, construction, and testing of fire protection systems are

Designator	Codes and Standards
	performed in accordance with the applicable portions of the NEPA codes, which invoke ANSI B31.1, American Water Worker Association (AWWA), American Petroleum Institute (API), and other codes, depending upon service. Quality assurance program requirements are implemented to ensure that the requirements for design, procurement, installation, testing, and administrative controls for the fire protection program are satisfied. The quality assurance requirements that apply to the fire protection program are described in Branch Technical Position APCSB 9.5-1, Appendix A.
C	Structure of structural components designed to codes and standards as defined in the design bases.
E	Electrical equipment designed to codes and standards as defined in the design bases.
J	Instrumentation and control equipment designed to codes and standards as defined in the design bases.

A more complete listing of applicable codes and standards is provided in Table 3.2-2.

TABLE 3.2-1 (Sheet 1 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
REACTOR COOLANT SYSTEM									
1. Reactor vessel and head	B	A	1	I	1	III	Y	Y	
2. Steam generator	B	A	1,2	I	1,2	III	Y	Y	
3. Pressurizer	B	A	1	I	1	III	Y	Y	
4. Pressurizer surge line	B	A	1	I	1	III	Y	Y	
5. Pressurizer relief lines (upstream of relief valves)	B	A	1	I	1	III	Y	Y	
6. Safety and relief valves	B	A	1	I	1	III	Y	Y	
7. Pressurizer relief tank	B	D	NNS	I	3	III	N	N	Note 1
8. Reactor coolant pump casing	B	A	1	I	1	III	Y	Y	
9. RCS loop piping	B	A	1	I	1	III	Y	Y	
10. Valves	B	A	1	I	1	III	Y	Y	
11. Reactor vessel supports	B	A	1	I	C	III-NF	Y	Y	
12. Steam generator supports	B	A	1	I	C	III-NF	Y	Y	
13. Pressurizer supports	B	A	1	I	C	III-NF	Y	Y	
14. Reactor coolant pump supports	B	A	1	I	C	III-NF	Y	Y	

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
15. Other safety-related supports and hangers	B	C	3	I	C	III-NF	Y	Y	
16. Lube oil drain tanks	B	D	NNS	II	4	API-650	N	N	
REACTOR SYSTEM									
1. Vessel internals	B	A	3	I	NG	III	Y	Y	
2. Fuel assemblies	B	NA	3	I	2	W	Y	Y	
3. Integrated head Package	B	C/D	3	I	NF	III	Y	Y	
4. CRDM-housing	B	A	1,2	I	1	III	Y	Y	
5. CRDM-mechanism	B	B	3	I	2	W	Y	Y	
6. Control rods	B	NA	3	I	2	W	Y	Y	
REACTOR HEAD VENT SYSTEM									
1. Piping and valves through second isolation	B	A	1	I	1	III	Y	Y	
2. All other piping and valves	B	B	2	I	2	III	Y	Y	
3. Instrumentation	B	NA	3	I	3	mfg	Y	Y	

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
SAFETY INJECTION SYSTEM									
1. Accumulators	B	B	2	I	2	III	Y	Y	
2. Boron injection tank	C	B	2	I	2	III	Y	Y	
3. Boron injection recirculation pumps	C	C	3	I*	3	III	Y	Y	
4. Boron recirculating pump motors	C	NA	3	I	E	NEMA MG1	Y	Y	
5. Boron injection surge tank	C	C	3	I	3	III	Y	Y	
6. Safety injection pumps	C	B	2	I	2	III	Y	Y	
7. Safety injection pump motors	C	NA	3	I	E	NEMA MG1	Y	Y	
8. Injection piping and valves downstream of check valves	B,C	A	1	I	1	III	Y	Y	
9. Accumulator discharge piping and valves downstream of MOVs	B	A	1	I	1	III	Y	Y	
10. Piping and valves from EWST and containment sumps to check valves	B	B	2	I	2	III	Y	Y	

TABLE 3.2-1 (Sheet 4 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
11. Piping from accumulators to MOVs	B	B	2	I	2	III	Y	Y	
12. Boron injection piping and valves downstream of check valve	B	A	1	I	1	III	Y	Y	
13. Boron injection piping and valves from charging pumps to check valve	C	B	2	I	2	III	Y	Y	
14. Safety-related instrumentation	B,C	NA	3	I	J	mfg	Y	Y	
15. Safety-related valve operators	B,C	NA	3	I	E	NEMA MG1	Y	Y	
RESIDUAL HEAT REMOVAL SYSTEM									
1. RHR pumps	C	B	2	I	2	III	Y	Y	
2. RHR pump motors	C	NA	3	I	E	NEMA MG1	Y	Y	
3. RHR HXs:									
Tube side, RHR	C	B	2	I	2	III TEMA-R	Y	Y	
Shell side, CCW	C	C	3	I	3	III TEMA-R	Y	Y	

TABLE 3.2-1 (Sheet 5 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
4. Piping and valves from hot legs to pump suction	B,C	A	1	I	1	III	Y	Y	
5. Piping and valves from pump suction injection check valves	C	B	2	I	2	III	Y	Y	
6. Safety-related instrumentation	B,C	NA	3	I	J	mfg	Y	Y	
CONTAINMENT SPRAY SYSTEM									
1. Containment spray pumps	C	B	2	I	2	III	Y	Y	
2. Containment spray pump motors	C	NA	3	I	E	NEMA MG1	Y	Y	
3. Spray nozzles	B	B	2	I	2	III	Y	Y	
4. Spray additive tank	C	C	3	I	3	III	Y	Y	
5. Spray eductors	C	B	2	I	2	III	Y	Y	

TABLE 3.2-1 (Sheet 6 of 42)

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
6. Process valves and piping from EWST, containment sumps, and educator inlet check valve to spray nozzles	B,C	B	2	I	2	III	Y	Y	
7. Process valves and piping downstream of spray additive tank to eductor inlet check valve	C	C	3	I	3	III	Y	Y	
8. Safety-related instrumentation	B,C	NA	3	I	J	mfg	Y*	Y	
CHEMICAL AND VOLUME CONTROL SYSTEM									
1. Volume control tank	C	B	2	I	2	III	Y	Y	
2. Boric acid storage tank	C	C	3	I	3	III	Y	Y	
3. Boric acid batching tank	C	D	NNS	II	C	API-650	N	N	
4. Boric acid transfer pumps	C	C	3	I	3	III	Y	Y	

TABLE 3.2-1 (Sheet 7 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
5. Boric acid transfer pump motors	C	NA	3	I	E	NEMA MG1	Y	Y	
6. Centrifugal charging pumps	C	B	2	I	2	III	Y	Y	
7. Centrifugal charging pump motors	C	NA	3	I	E	NEMA MG1	Y	Y	
8. Positive displacement	C	B	2	I	2	III	Y	Y	
9. Positive displacement charging pump motor	C	NA	3	II	E	NEMA MG1	N	N	
10. Regenerative HX	C	B	2	I	2	III TEMA-R	Y	Y	
11. Letdown HX:									
Tube side, CVCS	C	B	2	I	2	III TEMA-R	Y	Y	
Shell side, ACCW	C	D	NNS	I	C	III TEMA-R	N	N	Note n
12. Excess letdown HX:									
Tube side, CVCS	C	B	2	I	2	III TEMA-R	Y	Y	
Shell side, ACCW	C	D	NNS	I	C	III TEMA-R	N	N	Note n

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
13. CVCS cation bed demineralizers	C	C	3	I	3	III	Y	Y	
14. Mixed bed demineralizers	C	C	3	I	3	III	Y	Y	
15. Boric acid tank heater	C	NA	3	I	E	UL	Y	Y	
16. Chemical mixing tank	C	D	NNS	II	C	VIII	N	N	
17. Safety-related CVCS instrumentation	C	NA	3	I	J	mfg	Y	Y	
18. Piping to boric acid storage tank	C	C	3	I	3	III	Y	Y	
19. Piping and valves from boric acid storage tank	C	C	3	I	3	III	Y	Y	
20. Chemical mixing tank inlet and discharge piping	C	D	NNS	II	C	B31.1	N	N	
21. Piping and valves from reactor makeup water storage tank discharge	C	C	3	I	3	III	Y	Y	

TABLE 3.2-1 (Sheet 9 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
22. Letdown piping to volume control tank	C	B	2	I	2	III	Y	Y	
23. Charging piping from volume control tank	C	B	2	I	2	III	Y	Y	
24. Mixed bed demineralizer piping and valves	C	B/C	2,3	I	2,3	III	Y	Y	
25. Safety-related valve operators	C	NA	3	I	E	NEMA MG1	Y	Y	
BORON RECYCLE SYSTEM									
1. Recycle evaporator feed pumps	C	D	NNS	I	C	mfg	N	N	
2. Recycle evaporator feed pump motors	C	NA	3	II	E	NEMA MG1	N	N	
3. Boron recycle holdup tanks	C	C	3	I	3	III	Y	Y	
4. Recycle evaporator feed demin-eralizers	C	D	NNS	i	C	VIII	N	N	
5. Recycle evaporator condensate demineralizer	C	D	NNS	II	C	VIII	N	N	
6. Recycle evaporator package	C	D	NNS	I	C	VIII	N	N	

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
7. Recycle evaporator condensate filter	C	D	NNS	II	C	VIII	N	N	
8. Piping and valves	C	D	NNS	I,II	C	B31.1	N	N	
9. Valve operators	C	NA	3	I	E	NEMA MG1	N	N	
10. Instrumentation	C	NA	3	II	J	mfg	N	N	
CONTAINMENT ISOLATION SYSTEM									
1. Valves and piping	B,C	B	2	I	2	III	Y	Y	Selected valves and piping may be designed to ASME III-1
2. Valve operators	B,C	NA	3	I	E	NEMA MG1	Y	Y	
3. Instrumentation and controls	B,C	NA	3	I	J	mfg	Y	Y	
SERVICE WATER SYSTEM									
Not included in the NPB									
COMPONENT COOLING WATER SYSTEM									
1. CCW surge tanks	C	C	3	I	3	III	Y	Y	
2. CCW pumps	C	C	3	I	3	III	Y	Y	
3. CCW HXs	C	C	3	I	3	III TEMA-R	Y	Y	
4. CCW pump motors	C	NA	3	I	E	NEMA MG1	Y	Y	
5. CCW chemical addition tanks	C	D	NNS	II	C	API-650	N	N	

TABLE 3.2-1 (Sheet 11 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
6. Chemical addition tank valves and piping		D	NNS	II	C	B31.1	N	N	
7. All other process valves and piping		C	3	I	3	III	Y	Y	
8. Safety-related valve operators		NA	3	I	E	NEMA MG1	Y	Y	
9. Safety-related instrumentation		NA	3	I	J	mfg	Y	Y	
SPENT FUEL COOLING AND PURIFICATION SYSTEM									
1. SFP HXs	J	C	3	I	3	III TEMA-R	Y	Y	
2. SFP pumps	J	C	3	I	3	III	Y	Y	
3. SFP pump motors	J	NA	3	I	E	NEMA MG1	Y	Y	
4. Refueling water purification pump	J	D	NNS	II	C	mfg	N	N	
5. Refueling water purification pump motor	J	NA	3	II	E	NEMA MG1	N	N	
6. SFP demineralizer	J	NA	NNS	II	C	III	N	N	Note 1
7. SFP skimmer pump	J	NA	NNS	II	C	mfg	N	N	
8. SFP skimmer filter	J	NA	NNS	II	C	VIII	N	N	
9. Purification and skimmer-related valves and piping	J	NA	NNS	II	C	B31.1	N	N	

TABLE 3.2-1 (Sheet 12 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
10. Cooling-related valves and piping	J	C	3	I	3	III	Y	Y	
11. Safety-related instrumentation	J	NA	3	I	J	mfg	Y	Y	
12. Safety-related valve operators	J	NA	3	I	E	NEMA MG1	Y	Y	
REACTOR MAKEUP WATER SYSTEM									
1. Makeup pumps	J	D	NNS	I	C	III	N	N	Note 1
2. Makeup pump motors	J	NA	3	II	E	NEMA MG1	N	N	
3. Makeup water process piping and valves to SFP	J	D	NNS	I	C	III	N	N	Note 1
4. Instrumentation	J	NA	3	II	J	mfg	N	N	
5. All other piping	J	D	NNS	II	C	B31.1	N	N	
WASTE PROCESSING SYSTEM - LIQUID									Note q
1. Waste holdup tank	D	D	NNS	I	5	III	N	N	
2. Waste evaporator feed pump	D	D	NNS	I	5	III	N	N	
3. Waste evaporator feed pump motor	D	NA	3	II	E	NEMA MG1	N	N	
4. Waste evaporator	D	D	NNS	I	5	III	N	N	

TABLE 3.2-1 (Sheet 13 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
5. Waste evaporator concentrates holdup tank	D	D	NNS	II	5	API-650	N	N	
6. Waste evaporator concentrates holdup tank pump	D	D	NNS	II	5	mfg	N	N	
7. Waste evaporator reagent tank	D	D	NNS	II	C	VIII	N	N	
8. Waste evaporator condensate demineralizer	D	D	NNS	II	5	III	N	N	
9. Waste evaporator condensate pump	D	D	NNS	II	5	III	N	N	
10. Waste evaporator condensate tank	D	D	NNS	II	5	III	N	N	
11. Chemical drain tank	D	D	NNS	II	C	VIII	N	N	
12. Chemical drain tank pump	D	D	NNS	II	C	VIII	N	N	
13. Spent resin storage tank	D	D	NNS	I	5	III	N	N	
14. Waste monitor tank	D	D	NNS	II	5	III	N	N	
15. Waste monitor tank pump	D	D	NNS	II	5	III	N	N	
16. Waste monitor tank pump motor	D	NA	3	II	E	NEMA MG1	N	N	

TABLE 3.2-1 (Sheet 14 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
17. Laundry and hot shower tank	D	D	NNS	II	5	VIII	N	N	
18. Laundry and hot shower tank pump	D	D	NNS	II	5	mfg	N	N	
19. Reactor coolant drain tank	B	D	NNS	II	5	III	N	N	
20. Reactor coolant drain tank pump	B	D	NNS	II	5	III	N	N	
21. Piping and valves from waste evaporator and evaporator condensate demineralizer through evaporator condensate pump discharge valve	D	D	NNS	I,II	5	B31.1 III	N	N	
22. Instrumentation	D	NA	3	II	J	mfg	N	N	
WASTE PROCESSING SYSTEM - GASEOUS									
1. Gas decay tanks	D	C	3	I	3	III	Y	Y	
2. Gas decay tank drain pump	D	D	NNS	II	C	mfg	N	N	
3. Waste gas drain filter	D	D	NNS	II	C	VIII	N	N	

TABLE 3.2-1 (Sheet 15 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
4. Waste gas compressor package	D	C	3	I	3	III	Y	Y	
5. Process piping and valves	D	C	3	I	3	III	Y	Y	
6. Drain piping and valves	D	D	NNS	II	C	B31.1	N	N	
7. Safety-related valve operators	D	NA	3	I	E	NEMA MG1	Y	Y	
8. Safety-related instrumentation	D	NA	3	I	J	mfg	Y	Y	
RADWASTE VOLUME REDUCTION AND SOLIDIFICATION SYSTEM									
Not included in the NPB									
STEAM GENERATOR BLOWDOWN SYSTEM									
1. Blowdown HXs	J	D	NNS	II	5	VIII, TEMA-C	N	N	
2. Steam generator drain pump	J	D	NNS	II	5	mfg	N	N	
3. Demineralizers	J	D	NNS	II	5	VIII	N	N	
4. Spent resin storage tank	J	D	NNS	II	5	VIII	N	N	
5. Spent resin sluice pump	J	D	NNS	II	5	mfg	N	N	
6. Process valves and piping outside containment	J	D	NNS	II	5	B31.1	N	N	

TABLE 3.2-1 (Sheet 16 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
7. Process valves and piping inside containment through outer isolation valves	B	B	2	I	2	III	Y	Y	
8. Instrumentation	J	NA	3	II	J	mfg	N	N	
MAIN STEAM SYSTEM									
1. Steam generator (shell side)	B	B	2	I	2	III	Y	Y	
2. Piping from SG to 5-way restraint	B	B	2	I	2	III	Y	Y	
3. Safety valves	C	B	2	I	2	III	Y	Y	
4. Atmospheric relief valves	C	B	2	I	2	III	Y	Y	
5. Atmospheric relief valves operators	C	NA	3	I	E	NEMA MG1	Y	Y	
6. Main steam isolation valves	C	B	2	I	2	III	Y	Y	
7. Main steam isolation valve actuators	C	NA	3	I	E	mfg	Y	Y	
8. Safety-related valve operators	C	NA	3	I	E	NEMA MG1	Y	Y	

TABLE 3.2-1 (Sheet 17 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
9. Mechanical pressure, flow, and level instruments	B,C	B	2	I	2	III	Y	Y	
10. Safety-related instrumentation	B,C	NA	3	I	3	mfg	Y	Y	
11. Steam flow limiters	B	B	2	I	2	III	Y	Y	
EMERGENCY FEEDWATER SYSTEM									
1. Emergency feed pump turbine	C	C	3	I	3	III	Y	Y	
2. Emergency feed pumps	C	C	3	I	3	III	Y	Y	
3. Emergency feed motors	C	NA	3	I	E	NEMA MG[illegible]	Y	Y	
4. Piping up to SG feed admission MOVs	B,C	C	3	I	3	III	Y	Y	
5. Piping and valves including admission MOVs to SG	B,C	B	2	I	2	III	Y	Y	
6. Pump suction valves and piping	B,C	C	3	I	3	III	Y	Y	

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
7. Safety-related instrumentation	B,C	NA	3	I	J	mfg	Y	Y	
8. EFWS tank	B	C	3	I	3	III	Y	Y	
START-UP F.W. SYSTEM									
Not included in the NPB									
MAIN FEEDWATER SYSTEM									
1. Main feed line isolation valves	T	B	2	I	2	III	Y	Y	
2. Main feed line isolation valve motors	T	NA	3	I	E	NEMA MG1	Y	Y	
3. Main and auxiliary feed inlet check valves	T	B	2	I	2	III	Y	Y	
4. Piping	T	B	2	I	2	III	Y	Y	
5. Feed isolation instrumentation and controls	T	NA	3	I	J	mfg	Y	Y	
6. Safety-related valve motors	T	NA	3	I	E	NEMA MG1	Y	Y	
POWER CONVERSION SYSTEM									
Not included in the NPB									

TABLE 3.2-1 (Sheet 19 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
CONDENSER AIR EJECTOR SYSTEM									
					Not included in NPB				
CIRCULATING WATER SYSTEM									
					Not included in NPB				
INTAKE STRUCTURE SYSTEMS									
					Not included in NPB				
CIRCULATING WATER CHEMICAL INJECTION SYSTEM									
					Not included in NPB				
CONDENSATE CHEMICAL INJECTION SYSTEM									
					Not included in NPB				
CONDENSATE FILTER DEMINERALIZER SYSTEM									
					Not included in NPB				
PLANT MAKEUP WATER SYSTEM									
					Not included in NPB				

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
DEMINERALIZED WATER SYSTEM									
					Not included in NPB				
DIESEL GENERATOR SYSTEMS									
1. Day tanks	K	NA	3	I	C	III	Y	Y	
2. Fuel transfer pumps	K	C	3	I	3	III	Y	Y	
3. Fuel transfer pump motors	K	C	3	I	E	NEMA MG1	Y	Y	
4. Fuel filters	K	C	3	I	C	mfg	Y	Y	
5. Lube oil HX	K	C	3	I	C	III	Y	Y	
6. Lube oil heater	K	C	3	I	C	mfg	Y	Y	Casing is ASME VIII
7. Lube oil filters	K	C	3	I	C	III	Y	Y	
8. Cooling jacket water heater	K	NA	3	I	C	mfg	Y	Y	Casing is ASME VIII
9. Air compressors	K	C	3	II	C	mfg	N	N	
10. Air receivers	K	C	3	I	3	III	Y	Y	
11. Air compressor after coolers	K	C	3	II	C	mfg	N	N	
12. Safety-related instrumentation	K	NA	3	I	J	mfg	Y	Y	
13. Diesel generators	K	C	3	I	C,E	DEMA, NEMA	Y	Y	
14. Engine auxiliaries piping and valves	K	C	3	I	C	III	Y	Y	
15. Lube oil sump	K	C	3	I	C	III	Y	Y	
16. Fuel oil storage tanks	O	C	3	I	C	III	Y	Y	

TABLE 3.2-1 (Sheet 21 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
FIRE PROTECTION SYSTEMS									Note p
1. Diesel driven fire pumps	J	NA	NNS	II	7	NFPA	N	N	
2. Diesel engines	J	NA	NNS	II	7	mfg	N	N	
3. Motor driven fire pump	J	NA	NNS	II	7	NFPA	N	N	
4. Pump motors	J	NA	NNS	II	E	NEMA MG1	N	N	
5. Diesel fuel oil tanks	O	NA	NNS	II	7	NFPA	N	N	
6. Water storage tanks	O	NA	NNS	II	7	NFPA	N	N	
7. Water system piping and valves	VB	NA	NNS	II	7	NFPA	N	N	
8. Halon system piping, valves, and components	VB	NA	NNS	II	7	NFPA	N	N	
9. Instrumentation	VB	NA	NNS	II	J	mfg	N	N	
AUXILIARY GAS SYSTEMS (N_2, H_2, AND O_2)									
Not included in NPB									
RCS LIQUID SAMPLING SYSTEM									
1. Process sample valves and piping	C	D	NNS	II	4	B31.1	N	N	Note o
2. RCS sample valves and piping through outside containment isolation valve	B	B	2	I	2	III	Y	Y	

TABLE 3.2-1 (Sheet 22 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
3. Sample vessels	C	D	NNS	II	4	VIII	N	N	
4. Sample coolers	C	D	NNS	II	4	VIII	N	N	
5. Gross failed fuel detector	C	D	NNS	II	4	mfg	N	N	
6. Instrumentation	C	NA	NNS	II	J	mfg	N	N	
RCS GAS SAMPLING SYSTEM									
1. Sample vessels	C	D	NNS	II	4	VIII	N	N	
2. Valves and piping inside vent hood	C	D	NNS	II	4	B31.1	N	N	
3. Instrumentation	C	NA	NNS	II	J	mfg	N	N	
POST-ACCIDENT SAMPLING SYSTEM									
1. PASS panel (mechanical portions)	C	D	NNS	II	4	B31.1	N	N	
2. PASS panel instrumentation	C	NA	NNS	II	J	mfg	N	N	
3. Sample pumps	C	D	NNS	II	4	mfg	N	N	
4. Sample piping and valves	C	D	NNS	II	4	B31.1	N	N	
5. Containment penetration piping and valves	B	B	2	I	2	III	Y	Y	

TABLE 3.2-1 (Sheet 23 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
CONTAINMENT, AUXILIARY BUILDING, AND MISCELLANEOUS DRAIN SYSTEMS									
1. Radioactive drain sump pumps	D	D	NNS	II	4	mfg	N	N	
2. Reactor cavity sump pumps	B	D	NNS	I	4	mfg	N	N	
3. Containment sump pumps	B	D	NNS	I	4	mfg	N	N	
4. Penetration room sump pumps	C	D	NNS	I	4	mfg	N	N	
5. Auxiliary building sump pumps	J	D	NNS	II	4	mfg	N	N	
6. Component cooling water drain tank	C	D	NNS	II	4	API-650	N	N	
7. Component cooling water drain tank pump	C	D	NNS	II	4	mfg	N	N	
8. Diesel building oily waste sump pumps	K	D	NNS	II	4	mfg	N	N	
9. Equipment drain tank	C	D	NNS	II	4	API-650	N	N	
10. Floor drain tank	C	D	NNS	II	4	API-650	N	N	
11. Containment reactor cavity and penetration room sump pump motors	B,C	NA	NNS	I	E	mfg	N	N	
12. All other pump motors	C	NA	NNS	II	E	NEMA MG1	N	N	
13. Instrumentation	C	NA	NNS	II	J	mfg	N	N	
14. All other valves and all piping	C	D	NNS	II	4	B31.1	N	N	

TABLE 3.2-1 (Sheet 24 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
INSTRUMENT AND SERVICE AIR SYSTEM									
1. Air compressors	J	NA	NNS	II	C	mfg	N	N	
2. Air compressor motors	J	NA	NNS	II	E	NEMA MG1	N	N	
3. Air receivers	J	NA	NNS	II	C	VIII	N	N	
4. Aftercoolers	J	NA	NNS	II	C	VIII	N	N	
5. Moisture separators	J	NA	NNS	II	C	mfg	N	N	
6. Service air filters	J	NA	NNS	II	C	mfg	N	N	
7. Instrument air filters	J	NA	NNS	II	C	mfg	N	N	
8. Service air dryers	J	NA	NNS	II	C	mfg	N	N	
9. Instrument air dryers	J	NA	NNS	II	C	mfg	N	N	
10. Safety-related piping and valves (other than containment isolation)	VB	C	3	I	3	III	Y	Y	
11. Containment penetration piping and valves	B	B	2	I	2	III	Y	Y	
12. All other piping and valves	VB	NA	NNS	II	4	B31.1	N	N	
13. Instrumentation	VB	NA	NNS	II	J	mfg	N	N	
REACTOR COOLANT SYSTEM LEAK DETECTION SYSTEM									
1. Radiation monitors	B	NA	NNS	I	J	mfg	N	N	
2. Condensate measuring instruments	B	NA	NNS	II	J	mfg	N	N	

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
3. Condensate measuring piping and valves	B	D	NNS	II	4	B31.1	N	N	
4. Humidity detectors	B	NA	NNS	II	J	mfg	N	N	
5. Tank and sump level instruments	B	D	NNS	II	J	mfg	N	N	
FUEL HANDLING BUILDING DRAIN SYSTEM									
1. Sump pumps	F	D	NNS	II	4	mfg	N	N	
2. Sump pump motors	F	NA	NNS	II	E	NEMA MG1	N	N	
3. Piping and valves	F	D	NNS	II	4	B31.1	N	N	
4. Instrumentation	F	NA	NNS	II	J	mfg	N	N	
CONTAINMENT AIR COOLING SYSTEM									
1. Cooling fans	B	NA	3	I	C	AMCA	Y	Y	
2. Fan motors	B	NA	3	I	E	NEMA MG1	Y	Y	
3. Coolers	B	B	2	I	2	III	Y	Y	
4. Ductwork	B	NA	3	I	C	ANSI N509	Y	Y	
5. Dampers	B	NA	3	I	C	ANSI N509	Y	Y	
6. Damper motors	B	NA	3	I	E	NEMA MG1	Y	Y	
7. Safety-related instrumentation	B	NA	3	I	J	mfg	Y	Y	
CONTAINMENT PREACCESS FILTER SYSTEM									
1. Fans	B	D	NNS	II	C	AMCA	N	N	
2. Fan motors	B	NA	NNS	II	E	NEMA MG1	N	N	

TABLE 3.2-1 (Sheet 26 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
3. HEPA filters	B	D	NNS	II	C	ANSI N509	N	N	
4. Charcoal filters	B	D	NNS	II	C	ANSI N509	N	N	
5. Heaters	B	D	NNS	II	E	UL	N	N	
6. Moisture eliminators	B	D	NNS	II	C	ANSI N509	N	N	
7. Instrumentation	B	NA	NNS	II	J	mfg	N	N	
CONTAINMENT POST-LOCA PURGE EXHAUST SYSTEM									
1. Containment penetration ducting	B	B	2	I	2	III	Y	Y	
2. Isolation dampers	B	B	2	I	2	III	Y	Y	
3. Isolation damper motors	B	NA	3	I	E	NEMA MG1	Y	Y	
4. Heaters	C	NA	NNS	II	E	UL	N	N	
5. Moisture eliminators	C	D	NNS	II	C	ANSI N509	N	N	
6. HEPA filters	C	D	NNS	II	C	ANSI N509	N	N	
7. Charcoal filters	C	D	NNS	II	C	ANSI N509	N	N	
8. Post-LOCA purge exhaust unit housing	C	D	NNS	II	C	ANSI N509	N	N	
9. Instrumentation	C	NA	NNS	II	J	mfg	N	N	
CONTAINMENT CRDM COOLING SYSTEM									
1. Fans	B	NA	NNS	II	C	AMCA	N	N	
2. Fan motors	B	NA	NNS	II	E	NEMA MG1	N	N	
3. Ductwork	B	NA	NNS	II	C	SMACNA	N	N	
4. Instrumentation	B,C	NA	NNS	II	J	mfg	N	N	

TABLE 3.2-1 (Sheet 27 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
H_2 RECOMBINER AND MONITORING SYSTEM									
1. Penetration piping and valves	B	B	2	I	2	III	Y	Y	
2. Other piping and valves	C	C	3	I	3	III	Y	Y	
3. H_2 monitors	C	NA	3	I	J	mfg	Y	Y	
4. H_2 recombiners	C	NA	3	I	C	mfg	Y	Y	
5. Recombiner instrumentation	B,C	NA	3	I	J	mfg	Y	Y	
REACTOR EXTERNAL BUILDING HVAC SYSTEM									
1. Ventilation fans	C	NA	NNS	II	C	AMCA	N	N	
2. Ventilation fan motors	C	NA	NNS	II	E	NEMA MG1	N	N	
3. Ductwork	C	NA	NNS	II	C	ANSI N509	N	N	
4. Dampers	C	NA	NNS	II	C	ANSI N509	N	N	
5. Damper motors	C	NA	NNS	II	E	NEMA MG1	N	N	
6. Instrumentation	C	NA	NNS	II	J	mfg	N	N	
CONTROL ROOL HVAC SYSTEM (ESSENTIAL PORTION)									
1. Filter unit fans	G	NA	3	I	C	AMCA	Y	Y	
2. Filter unit fans motors	G	NA	3	I	E	NEMA MG1	Y	Y	
3. Return air fans	G	NA	3	I	C	AMCA	Y	Y	
4. Return air fans motor	G	NA	3	I	E	NEMA MG1	Y	Y	

TABLE 3.2-1 (Sheet 28 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
5. Moisture eliminators	G	NA	3	I	C	ANSI N509	Y	Y	
6. Heaters	G	NA	3	I	E	UL	Y	Y	
7. HEPA filters	G	NA	3	I	C	ANSI N509	Y	Y	
8. Charcoal filters	G	NA	3	I	C	ANSI N509	Y	Y	
9. Cooling coils	G	C	3	I	3	III	Y	Y	
10. Dampers	G	NA	3	I	C	ANSI N509	Y	Y	
11. Damper motors	G	NA	3	I	E	NEMA MG1	Y	Y	
12. Ductwork	G	NA	3	I	C	ANSI N509	Y	Y	
13. Safety-related instrumentation	G	NA	3	I	J	mfg	Y	Y	
ESSENTIAL SAFETY FACILITY AREA HVAC SYSTEM									
1. Battery room exhaust fans	I	NA	3	I	C	AMCA	Y	Y	
2. Battery room exhaust fan motors	I	NA	3	I	E	NEMA MG1	Y	Y	
3. A/C unit fans	I	NA	3	I	C	AMCA	Y	Y	
4. A/C unit fan motors	I	NA	3	I	E	NEMA MG1	Y	Y	
5. Cooling coils	I	C	3	I	3	III	Y	Y	
6. Prefilters	I	NA	3	I	C	ASHRAE	Y	Y	
7. Ductwork	I	NA	3	I	C	ANSI N509	Y	Y	
8. Dampers	I	NA	3	I	C	ANSI N509	Y	Y	
9. Damper motors	I	NA	3	I	E	NEMA MG1	Y	Y	
10. Safety-related instrumentation	I	NA	3	I	J	mfg	Y	Y	

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
CONTROL COMPLEX AREA CABLE SPREADING ROOM HVAC SYSTEM									
1. Cable spreading room A/C fans	G	NA	NNS	II	C	AMCA	N	N	
2. Cable spreading room A/C fan motors	G	NA	NNS	II	E	NEMA MG1	N	N	
3. Prefilters	G	NA	NNS	II	C	ASHRAE	N	N	
4. All other coolers	G	NA	NNS	II	C	ARI	N	N	
5. Electric duct heaters	G	NA	NNS	II	E	UL	N	N	
6. All other ductwork	G	NA	NNS	II	C	SMACNA	N	N	
7. All other instrumentation	G	NA	NNS	II	J	mfg	N	N	
FUEL HANDLING AREA POST-ACCIDENT EXHAUST SYSTEM									
1. Fans	F	NA	3	I	C	AMCA	Y	Y	
2. Fan motors	F	NA	3	I	E	NEMA MG1	Y	Y	
3. Moisture eliminators	F	NA	3	I	C	ANSI N509	Y	Y	
4. Heaters	F	NA	3	I	E	UL	Y	Y	
5. HEPA filters	F	NA	3	I	C	ANSI N509	Y	Y	
6. Charcoal filters	F	NA	3	I	C	ANSI N509	Y	Y	
7. Ductwork	F	NA	3	I	C	ANSI N509	Y	Y	
8. Dampers	F	NA	3	I	C	ANSI N509	Y	Y	
9. Dampers motors	F	NA	3	I	E	NEMA MG1	Y	Y	
10. Safety-related instrumentation	F	NA	3	I	J	mfg	Y	Y	

TABLE 3.2-1 (Sheet 30 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
REACTOR EXTERNAL BUILDING OUTSIDE AIR SUPPLY, NORMAL HVAC, RADIOACTIVE FILTER EXHAUST, AND CONTINUOUS EXHAUST SYSTEMS									
1. Exhaust unit fans	C	NA	NNS	II	C	AMCA	N	N	
2. Exhaust unit fan motors	C	NA	NNS	II	E	NEMA MG1	N	N	
3. A/C fans	C	NA	NNS	II	C	AMCA	N	N	
4. A/C fan motors	C	NA	NNS	II	E	NEMA MG1	N	N	
5. HEPA filters	C	NA	NNS	II	C	ANSI N509	N	N	
6. Charcoal filters	C	NA	NNS	II	C	ANSI N509	N	N	
7. Electric heaters	C	NA	NNS	II	E	UL	N	N	
8. Cooling coils	C	NA	NNS	II	C	ARI	N	N	
9. Ductwork	C	NA	NNS	II	C	SMACNA	N	N	
10. Dampers	C	NA	NNS	II	C	ANSI N509	N	N	
11. Damper motors	C	NA	NNS	II	E	NEMA MG1	N	N	
12. Instrumentation	C	NA	NNS	II	J	mfg	N	N	
PIPING PENETRATION VENTILATION SYSTEM									
1. Restraint cooling fans	C	NA	NNS	II	C	AMCA	N	N	
2. Restraint cooling fan motors	C	NA	NNS	II	E	NEMA MG1	N	N	
3. Ductwork	C	NA	NNS	II	C	SMACNA	N	N	
4. Backdraft dampers	C	NA	NNS	II	C	ANSI N509	N	N	
5. Instrumentation	C	NA	NNS	II	J	mfg	N	N	
PIPING PENETRATION FILTER EXHAUST SYSTEM									
1. Fans	C	NA	3	I	C	AMCA	Y	Y	
2. Fan motors	C	NA	3	I	E	NEMA MG1	Y	Y	

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
3. HEPA filters	C	NA	3	I	C	ANSI N509	Y	Y	
4. Charcoal filters	C	NA	3	I	C	ANSI N509	Y	Y	
5. Dampers	C	NA	3	I	C	ANSI N509	Y	Y	
6. Damper motors	C	NA	3	I	E	NEMA MG1	Y	Y	
7. Ductwork	C	NA	3	I	C	ANSI N509	Y	Y	
8. Area coolers	C	C	3	I	3	III	Y	Y	
9. Safety-related instrumentation	C	NA	3	I	J	mfg	Y	Y	
ELECTRICAL PENETRATION FILTER EXHAUST SYSTEM									
1. Fans	C	NA	3	I	C	AMCA	Y	Y	
2. Fan motors	C	NA	3	I	E	NEMA MG1	Y	Y	
3. HEPA filters	C	NA	3	I	C	ANSI N509	Y	Y	
4. Charcoal filters	C	NA	3	I	C	ANSI N509	Y	Y	
5. Ductwork	C	NA	3	I	C	ANSI N509	Y	Y	
6. Dampers	C	NA	3	I	C	ANSI N509	Y	Y	
7. Damper motors	C	NA	3	I	E	NEMA MG1	Y	Y	
8. Safety-related instrumentation	C	NA	3	I	J	mfg	Y	Y	
DIESEL GENERATOR BUILDING HVAC SYSTEM									
1. ESF exhaust fans	K	NA	3	I	C	AMCA	Y	Y	
2. ESF exhaust fan motors	K	NA	3	I	E	NEMA MG1	Y	Y	
3. ESF ductwork	K	NA	3	I	C	ANSI N509	Y	Y	

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
4. ESF dampers	K	NA	3	I	C	ANSI N509	Y	Y	
5. ESF damper motors	K	NA	3	I	E	NEMA MG1	Y	Y	
6. ESF instrumentation	K	NA	3	I	J	mfg	Y	Y	
MAIN CONTROL BOARD									Note j, k
1. Hand switches and controls for safety-related equipment	G	NA	3	I	J	mfg	Y	Y	
2. All other instruments and controls	G	NA	6	I	J	mfg	N	N	
NUCLEAR INSTRUMENTATION SYSTEM									
1. All instruments inputting to reactor protection system	B,C	NA	3	I	J	mfg	Y	Y	
PROCESS CONTROL SYSTEM									
1. NSSS safety-related instrumentation and controls	B,C	NA	3	I	J	mfg	Y	Y	

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
PROTECTION SYSTEM									
1. Protection instrumentation and controls	B,C	NA	3	I	J	mfg	Y	Y	
ROD CONTROL POWER SYSTEM									
1. Reactor trip switchgear	I	NA	3	I	E	mfg	Y	Y	
2. Other switchgear	I	NA	NNS	II	E	mfg	N	N	
ROD CONTROL SYSTEM									
1. Rod control equipment	I	NA	NNS	I	J	mfg	N	N	
ROD POSITION INDICATION SYSTEM									
1. Rod position instrumentation	I	NA	NNS	I	J	mfg	N	N	
RADIATION MONITORING SYSTEM									
1. Safety-related portions	VB	NA	3	I	J	mfg	Y	Y	

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
2. Nonsafety-related, seismic Category I portions	VB	NA	NNS	I	J	mfg	N	N	
3. Other portions	VB	NA	NNS	II	J	mfg	N	N	
ESF ACTUATION SYSTEM									
1. All portions	B,C	NA	3	I	J	mfg	Y	Y	
REACTOR INSTRUMENTATION									
1. All portions inputting to reactor protection	B,C	NA	3	I	J	mfg	Y	Y	
2. Other portions	B,C	NA	NNS	II	J	mfg	N	N	
REACTOR CONTROL SYSTEM									
1. Protection-related portions	B,C	NA	3	I	J	mfg	Y	Y	
2. Other portions	B,C	NA	NNS	II	J	mfg	N	N	
POST-ACCIDENT MONITORING SYSTEM									Note k
1. Safety-related portions	B,C	NA	3	I	J	mfg	Y	Y	

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
2. Nonsafety-related, seismic Category I portions	B,C	NA	NNS	I	J	mfg	N	N	
3. Other portions	B,C	NA	NNS	II	J	mfg	N	N	
PLANT AUXILIARY CONTROL BOARDS									
1. Safety-related portions	C	NA	3	I	J	mfg	Y	Y	
2. Nonsafety-related portions	C	NA	NNS	II	J	mfg	N	N	
INCORE INSTRUMENTATION									
1. All portions	B,C	NA	NNS	I	J	mfg	N	N	
COMPUTER SYSTEM									
1. All portions	VB	NA	NNS	I	J	mfg	N	N	
ANNUNCIATOR SYSTEM									
1. All portions	VB	NA	NNS	I	J	mfg	N	N	
SEISMIC MONITORING EQUIPMENT									
1. All portions	VB	NA	NNS	I	J	mfg	N	N	

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
AC SYSTEM - 480 V									
1. 4160/480 V transformers	N	NA	3	I	E	mfg	Y	Y	
2. Load centers	C	NA	3	I	E	mfg	Y	Y	
3. Motor control centers	C	NA	3	I	E	mfg	Y	Y	
4. Instrumentation and control	C	NA	3	I	E	mfg	Y	Y	
AC SYSTEM - 4160 V									
1. 4.16 kV buses and switchgear	C	NA	3	I	E	mfg	Y	Y	
2. Instrumentation and controls	C	NA	3	I	E	mfg	Y	Y	
DC SYSTEM - CLASS 1E									
1. Batteries	C	NA	3	I	E	mfg	Y	Y	
2. Chargers	C	NA	3	I	E	mfg	Y	Y	
3. Breakers, bus work, and switchgear	C	NA	3	I	E	mfg	Y	Y	
4. Instrumentation and controls	C	NA	3	I	E	mfg	Y	Y	
5. Motor control center	C	NA	3	I	E	mfg	Y	Y	
6. Distribution panels	C	NA	3	I	E	mfg	Y	Y	

TABLE 3.2-1 (Sheet 37 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
120-V ac POWER SYSTEM - CLASS 1E									
1. Transformers	C	NA	3	I	E	mfg	Y	Y	
2. Breakers, bus work, and switchgear	C	NA	3	I	E	mfg	Y	Y	
3. dc-ac inverters	C	NA	3	I	E	mfg	Y	Y	
4. Instrumentation and control	C	NA	3	I	E	mfg	Y	Y	
5. Distribution panels	C	NA	NNS	II	E	mfg	N	N	
AC SYSTEM - 13.8 kV									
1. 13.8-kV buses and switchgear	C	NA	NNS	II	E	mfg	N	N	
2. 13.8-kV RCP 1E breakers	C	NA	3	I	E	mfg	Y	Y	
STRUCTURES AND BUILDINGS									
1. Containment building	B	B	2	I	MC	ASME III,	Y	Y	
2. Equipment hatch and personnel locks	B	B	2	I	MC	III	Y	Y	
3. Penetration sleeves assemblies	B	B	2	I	MC	III	Y	Y	

TAL.E 3.2-1 (Sheet 38 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
4. Fuel transfer tube housing and bellows assembly	C	C	3	I	MC	III	Y	Y	
5. Reactor external building	C	NA	3	I	C	AISC ACI 318	Y	Y	
6. Diesel generator area building	K	NA	3	I	C	AISC ACI 318	Y	Y	
7. Fuel handling area	F	NA	3	I	C	AISC ACI 318	Y	Y	
8. Control complex area	G	NA	3	I	C	AISC ACI 318	Y	Y	
9. Diesel fuel oil storage tanks	O	NA	3	I	C	API-650	Y	Y	
10. Spent fuel pool and refueling canal liner plate	F	NA	3	I	C	AISC ACI 318	Y	Y	
11. Containment internal structures	B	NA	3	I	C	AISC ACI 318 ACI349 III-NF	Y	Y	
12. Electrical cable trays and supports	VB	NA	3	I	C	AISC	Y	Y	
13. Pipe supports	VB	NA	3	I	C	AISC	Y	Y	
14. Pipe whip restraints	VB	NA	3	I	C	AISC	Y	Y	

TABLE 3.2-1 (Sheet 39 of 42)
CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Principal System and Components	(a) Location	(b) Quality Group	(c) Safety Class	(d) Seismic Category	(e) Code Classification	(f) Principal Construction Code & Stds.	(g) Q-List	(h) Safety Related	(i) Comments
FUEL HANDLING SYSTEM									
1. New and spent fuel storage racks	F	NA	NNS	I	C	mfg	N	N	
2. Refueling machine	B	NA	NNS	II	C	mfg	N	N	
3. RCC storage station	B	NA	NNS	II	C	mfg	N	N	
4. Thimble plug storage rack	B	NA	NNS	II	C	mfg	N	N	
5. Integrated head cable assembly	B	NA	NNS	II	C	mfg	N	N	
6. Integrated head cable tray	B	NA	NNS	II	C	mfg	N	N	
7. Integrated head lifting rig	B	NA	NNS	II	C	mfg	N	N	
8. Integrated head missile shield	B	C	3	I	NF	III-NF	Y	Y	
9. Reactor internals lifting rig	B	NA	NNS	II	C	mfg	N	N	
10. Fuel handling machine	P	NA	NNS	II	C	mfg	N	N	
16. Spent fuel handling tool	F	NA	NNS	II	C	mfg	N	N	
17. New fuel handling tool	F	NA	NNS	II	C	mfg	N	N	
18. Fuel transfer tube	F	C	3	I	MC	III	Y	Y	
19. Fuel transfer system	F	NA	NNS	II	C	mfg	N	N	

TABLE 3.2-1 (Sheet 40)

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

FOOTNOTES AND COMMENTS

a. Location

A	Reactor Building (B,C)*
B	Containment*
C	Reactor External Building (F,G,H,I,J,K)*
F	Fuel Handling Area*
G	Control Complex Area*
H	Main Steam Tunnel*
I	Essential Safety Facility Area*
J	Auxiliary Equipment Area*
K	Diesel Generator Area or Building*
D	Waste Disposal building
N	Transformer Area
P	Switch Yard
T	Turbine Building
VB	Various Buildings
O	Outside
*	Nuclear Power Block

b. Quality Group

The quality group classification corresponds to those provided in Regulatory Guide 1.26. NA indicates not applicable and is used for equipment and structures that do not fall under the purview of Regulatory Guide 1.26.

c. Safety Class

The nuclear safety class corresponds to the quality group classifications in ANSI/ANS-51.1-1983, Nuclear Safety Classes 1, 2, 3, and NNS.

TABLE 3.2-1 (Sheet 41)

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

d. Seismic Category

Seismic Category I is applied to those safety-related structures, systems, and components that must remain functional during and after a safe shutdown earthquake (SSE) according to Regulatory Guide 1.29 and to those nonsafety-related structures, systems, and components that are designed to Seismic Category I requirements.

e. Code, Classification

See Subsection 3.2.2.3.

f. Principal Construction Code

The codes referenced are primary codes only and are defined in Table 3.2-2. Detailed construction codes are listed in the component specification.

g. Q-List

Y Yes; requires compliance with 10 CFR 50, Appendix B, as implemented in the WAPWR quality assurance programs.

N No; not within the scope of 10 CFR 50, Appendix B.

h. Safety Related

Y Yes; safety related.
N No; not safety related.

i. Comments

This column contains a listing of appliable design criteria and other amplifying information.

TABLE 3.2-1 (Sheet 42)

CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

j. The main control board will be qualified by Westinghouse to Institute of Electrical and Electronics Engineers (IEEE) 344 and 323.

k. Post-accident monitoring system (PAMS) instruments are assigned a classification based on their category as defined in Regulatory Guide 1.97, Revision 2.

l. Selected materials, components, parts, appurtenances, and piping subassemblies are procured in accordance with ASME Code, Section III, Class 3; however, the system is designed and installed in accordance with ANSI B31.1. Conformance with these aspects of the ASME code is required only for initial procurement.

n. The entire heat exchanger is constructed to ASME Section III requirements to ensure the integrity of the safety-related portion.

o. Portions of the sample system that are part of the pressure boundary of the system being sampled must meet the same quality and code requirements as that sampled system up to and including the first normally shut isolation valve in the sample line.

p. The quality assurance program to be applied to fire protection systems is described in Branch Technical Position APCSB 9.5-1, Appendix A, attached to Nuclear Regulatory Commission (NRC) Standard Review Plan 9.5.1.

q. The quality assurance program to be applied to radioactive waste management systems is described in Regulatory Guide 1.143.

r. Heat tracing and associated heat tracing equipment for the safety injection system is redundant, procured as Class 1E, and seismically and environmentally qualified in accordance with IEEE 323, 344, and 383.

TABLE 3.2-2 (Sheet 1 of 3)

PRINCIPAL CODES AND STANDARDS

I	ASME Boiler and Pressure Vessel Code, Section I.
III-1,2,3, MC, NF, CS	ASME Boiler and Pressure Vessel Code, Section III, Class 1, 2, 3, or MC, NF, or CS.
VIII	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1.
B31.1	ANSI B31.1.0, Power Piping.
AISC	American Institute of Steel Construction.
AISI	American Iron and Steel Institute, Specification for the Design of Cold-Formed Steel Structural Members, 1968, Design of Light Gage Cold-Formed Stainless Steel Structural Members.
AMCA	Air Moving and Conditioning Association.
ACI 318	American Concrete Institute, Building Code Requirements for Reinforced Concrete.
ACI-349	American Concrete Institute, Code Requirements for Nuclear Safety Related Structures.
ANSI N509	American National Standard Institute, Nuclear Power Plant Air Cleaning Units and Components, 1976.
API-620	American Petroleum Institute, Recommended Rules for Design and Construction of Large, Low Pressure Storage Tanks.

TABLE 3.2-2 (Sheet 2 of 3)

PRINCIPAL CODES AND STANDARDS

API-650	American Petroleum Institute, Welded Steel Tanks for Oil Storage.
ARI	Air Conditioning and Refrigeration Institute.
ASHRAE	American Society of Heating, Refrigerating and Air Conditioning Engineers.
AWWA	American Water Works Association
DEMA	Diesel Engine Manufacturer Association, Standard Practices for Stationary Diesel and Gas Engines, 1971.
IEEE	Institute of Electrical and Electronic Engineers
mfg	Manufacturer's standard. Design requirements specified by designer with appropriate consideration of the intended service and operating conditions.
NEMA MG1	National Electric Manufacturers Association, 1972, Motors and Generators.
NFPA	National Fire Protection Association.
SMACNA	Sheet Metal and Air Conditioning Contractors National Association, Inc.
TEMA C,R	Tubular Exchanger Manufacturers Association, Class C or R.

TABLE 3.2-2 (Sheet 3 of 3)

PRINCIPAL CODES AND STANDARDS

UBC	Uniform Building Code.
UL	Underwriters' Laboratories.

3.3 WIND AND TORNADO LOADINGS

3.3.1 Wind Loadings

All structures are designed for wind loading in accordance with American National Standards Institute (ANSI) A58.1-1982 "Minimum Design Loads in Buildings and Other Structures" (Reference 1).

3.3.1.1 Design Wind Velocity

The design wind is specified as a basic wind speed of 110 mph with an annual probability of occurrence of 0.02. This wind speed is the fastest mile wind speed at 10 meters above the ground in open terrain (ANSI A58.1 Exposure C). The magnitude of 110 mph has been selected based on the most severe location identified in ANSI A58.1. The annual probability of occurrence of 0.02 is the basis established in ANSI A58.1 for the basic wind speed. Higher winds with a probability of occurrence of 0.01 are considered in the design by utilizing an Importance Factor of 1.11. This is obtained by classifying the NPB as an essential facility at a hurricane oceanline and using the design provisions for Category III of ANSI A58.1.

Vertical velocity profiles and gust response factors are calculated in accordance with ANSI A58.1 for Exposure C.

3.3.1.2 Determination of Applied Forces

Effective pressures applied to interior and exterior surfaces of the buildings and the corresponding shape coefficients are calculated in accordance with ANSI A58.1 for Exposure C. Shape coefficients for the reactor exterior building are calculated using ASCE paper 3269, Reference 2.

3.3.2 Tornado Loadings

Seismic Category I structures are designed to resist tornado loads without exceeding the allowable stresses defined in Subsection 3.8.4. In addition, all Seismic Category I structures are designed to remain functional when subjected to wind generated missiles. Seismic Category I structures may sustain local missile damage such as partial penetration and local cracking and/or permanent deformation, provided that structural integrity is maintained and Seismic Category I systems, components and equipment required to function during or after passage of a tornado are not subject to damage by secondary missiles, such as from concrete spalling.

3.3.2.1 Applicable Design Parameters

The tornado used in the design of the NPB is the tornado specified in ANSI/ANS 2.3-1983 "Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites." Since the nuclear power block is intended for a wide range of sites, the maximum windspeed is selected at 320 mph which is the maximum specified corresponding to a probability of 10^{-7} per year for any location in the U.S.

The ANSI/ANS 2.3 standard is based on detailed analyses and evaluation of the data by experts leading to issue of the concensus standard in 1983. It represents more recent in-depth evaluation than was incorporated in Regulatory Guide 1.76 and the Standard Review Plan.

The design parameters appl'cable to the design basis tornado are as follows:

- o Maximum wind speed - 320 mph
- o Translational speed - 70 mph maximum/5 mph minimum
- o Radius from the center of the tornado, where the maximum wind velocity occurs - 540 ft.
- o Atmospheric pressure drop - 1.96 psi.

3.3.2.2 Determination of Forces on Structures

The procedures specified in Subsection 3.3.1.2 are used to transform the tornado wind loading and differential pressure loading into effective loads on structures, with a wind velocity of 320 mph (translational plus rotational velocities). The dynamic wind pressure is applied to the structure in the same manner as the wind loads described in Subsection 3.3.1.2, with the exception that the importance factor, gust factor and the variation of wind speed with height do not apply. Loading combinations and load factors used are as follows:

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

where

W_t = total tornado load
W_w = total wind load
W_p = total differential pressure load
W_m = total missile load

The maximum pressure drop of 1.96 psi, applicable to a nonvented structure, is used for W_p, unless a lower value is justified using the provisions of Reference 3 for partially vented structures. When the tornado loading includes the missile load, the structure locally may go in the plastic range due to the missile impact.

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

The failure of structures, not designed for tornado loadings will not affect the capability of Category I structures or systems performance. This will be accomplished by any one of the following:

a. Designing the adjacent structure to Category I structure tornado loading.

b. Investigating the effect of adjacent structural failure on Category I structures to determine that no impairment of function results.

c. Designing a structural barrier to protect Category I structures from adjacent structural failure.

3.3.3 Design and Analysis Procedures

The pressure due to normal winds or tornado winds will be considered static and applied to the structure like any other conventional design loading. The tornado loads will include the tornado wind pressure, internal pressure due to tornado created atmospheric pressure drop, and forces due to tornado-generated missiles. The normal wind loads or tornado wind loads will be combined with other loads as described in Subsection 3.8.4. Conventional design techniques will be utilized to analyze and design the structures for these loadings in the same manner as for other static loadings.

3.3.4 References

1. ANSI A58.1, "American National Standard Minimum Design Loads for Buildings and Other Structures," A58.1-1982, American National Standards Institute (Revision of ANSI A58.1-1972).

2. ASCE Paper No. 3269, "Wind Forces on Structures," Transactions of the American Society of Civil Engineers, Vol. 126, Part II (1961).

3. U.S. Nuclear Regulatory Commission, Standard Review Plan, 3.3.2: "Tornado Loadings," SRP 3.3.2 Rev. 2, July 1981

3.4 WATER LEVEL (FLOOD) DESIGN

This section evaluates the Nuclear Power Block (NPB) portions of the WAPWR for effects of water level (floods). A listing of those WAPWR structures, systems, and components within the scope of the NPB is presented in Table 3.2-1 of this module.

The flooding of a nuclear power plant from natural causes can be attributed to probable maximum flood (PMF), site and adjacent area probable maximum precipitation (PMP) runoff, and ground water. Site interface parameters for flood level and PMP are listed in Table 1.9-3 of RESAR-SP/90 PDA Module 3.

NPB Seismic Category I structures, systems, and components whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity are protected from the effects of the design basis flood levels or flood conditions including wave and wind effects by the following methods:

a) Designed to withstand effects of the design basis flood level or flood condition.

b) Positioned to preclude effects of the design basis flood level or flood condition.

c) Housed within structures which satisfy method "a" or "b" above.

Criteria for the design basis flood conform to the guidelines of Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants" and Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," as well as meeting the relevant requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Section IV.C as related to protecting structures, systems, and components important to safety from the effects of floods, tsunamis and seiches.

3.4.1 Flood Protection

This section discusses the flood protection measures provided for NPB Category I structures, systems, and components.

3.4.1.1 External Flood Protection

3.4.1.1.1 Structural Flood Protection

All NPB plant structures and systems they house are designed to withstand the effects of flooding. Systems located above grade are protected from flooding by maintaining the elevation of doors and openings in the exterior walls of the structures above the final grade elevation. This provides protection from flooding due to ponding of surface water. Waterstops and waterproofing materials are not used on structures above grade since the concrete walls of these structures provide adequate waterproofing during periods of flooding caused by heavy precipitation.

NPB systems located below grade are protected by a combination of a waterproofing system for the structures and the location of safety related systems in watertight compartments. In addition, an interior floor drainage system is provided within the structures.

Waterproofing is provided below grade by means of waterstops and waterproofing materials. Waterstops are provided at expansion and construction joints of walls and basemats located below grade. Waterstop material is of synthetic rubber.

The exterior waterproofing system is applied to the vertical exterior surfaces of walls below grade and the bottom surface of basemats of NPB plant structures. The exterior waterproofing system consists of a membrane composed of several coats of asphalt reinforced with alternate layers of asphalt-coated glass fabric. The number of coats is consistent with the design hydrostatic head in accordance with the manufacturer's recommendations.

Below-grade penetrations are provided with waterproof seals.

3.4.1.1.2 Surface Drainage System

The surface drainage system is not within the scope of the NPB.

3.4.1.2 Flood Protection for Flooding from Component Failures

The basis for postulating piping failures is described in Section 3.6.

The following design features have been incorporated to meet the criteria of Subsection 3.6.1:

a) Redundant safety related components are either located in separate compartments, protected from flooding by adequate separation, or protected from flooding by natural drainage.

b) Watertight rooms used to protect safety related equipment from damage from flooding have watertight access doors fitted with switches and circuits that provide an alarm in the control room when the access door is open. The watertight access doors are designed to withstand the water pressure exerted by the calculated flood levels.

c) Passages or piping and other penetrations through walls of a room containing equipment important to safety are sealed against water leakage from any postulated failure of water systems.

d) Walls, doors, panels, or other compartment closures designed to protect equipment important to safety from damage due to flooding from a system rupture are designed for effects of the system rupture.

e) Rooms containing system components and pipes whose rupture could result in flood damage to equipment important to safety have level alarms that alarm in the control room.

f) Equipment is either located or protected such that rupture of a system connected to a body of water (ocean, reservoir, etc.) will not result in failure of other essential equipment from flooding.

Each area of the plant is reviewed to determine the postulated fluid system failure, including non-Seismic Category I and non-tornado protected tanks, vessels, and other process equipment, which results in the most adverse flooding conditions. Flooding levels are determined for various areas. Included are consideration of a component failure in the circulating water system as well as actuation of the fire protection system outside containment. The levels calculated are not sufficient to impair either the operability of essential systems and components or damage essential structures. The containment and all essential equipment located within are designed to withstand the environment associated with the DBA (LOCA or MSLB) inside containment. These environmental conditions consider the maximum resultant flooding level associated with the rupture of the largest reactor coolant system pipe.

The safety related equipment is housed inside Seismic Category I structures and thus is protected against flooding. The safety related structures will not be jeopardized as a result of the maximum still water level or wave run-up resulting from a PMF, or storm water accumulated at the plant site due to a PMP, and therefore, it will not be necessary to bring the reactor to a cold shutdown for flood conditions.

3.4.1.3 Permanent Dewatering System

A permanent dewatering system is not within the scope of the NPB.

3.4.2 Analysis Procedures

3.4.2.1 Analysis Procedures for External Flooding

The NPB Seismic Category I structures are designed to protect the safety related systems, equipment, and components from the probable maximum flood and the highest groundwater level. These structures meet the requirements of GDC 2 with respect to their capability to withstand the effects of the flood or highest groundwater level so that their design reflects:

1. appropriate consideration for the most severe flood recorded for the site with an appropriate margin,

2. appropriate combination of the effects of normal and accident conditions with the effect of the natural phenomena, and

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

3.4.2.2 Analysis Procedures for Flooding from Component

The effects of major flooding as a result of component failures or actuation of the fire protection system outside containment are determined. The maximum expected flood level inside the containment as a result of a large break LOCA is calculated. All safety related equipment required to mitigate the effects of the LOCA are located at or above this elevation inside the containment.

Each postulated failure in liquid-carrying system piping is considered separately as a single postulated event occurring during normal plant operation. Each area of the plant is reviewed to determine the failure which results in the most adverse flooding conditions.

The type of pipe failures considered are:

a) High-energy line break

b) Moderate-energy pipe breaks

c) Moderate-energy pipe cracks

d) Fire water system actuation for 10 minutes followed by discharge from two fire hoses for 10 minutes.

The flow from the postulated failure is assumed to result in a flood in the compartment in which the component is located, except that consideration is given to unprotected communicating compartments. The volume occupied by equipment in a room is considered negligible except where it is apparent that large equipment occupies a significant proportion of the available room volume. Examples of this are rooms specifically designed to accommodate large storage tanks.

The analysis identifies high-energy fluid piping failure which could lead to unacceptable flooding conditions. Flood protection design features, as discussed in Subsection 3.4.1.1, are implemented to mitigate these consequences thus eliminating potentially unacceptable flooding conditions. The effects of pipe whip and jet impingement from high-energy piping are not considered in this criteria except where the effects may give rise to sources of further flooding.

All safety related structures are designed for the effect of ground water buoyant forces. The electrical manholes for NPB auxiliary and emergency power system cables are also Seismic Category I reinforced concrete structures, but are founded on soil. Electrical manholes and buried duct runs for NPB auxiliary and emergency power system cables are capable of normal function while completely or partially flooded. The duct runs are sloped towards the electrical manholes and groundwater finding its way into the conduit will be drained to the electrical manhole. The electrical manholes are provided with collection sumps for any water coming through conduits or cracks in the reinforced concrete walls or slabs of the manholes. When necessary, the water in the sumps will be removed by portable pumps connected to the pipes from the sumps.

3.5 MISSILE PROTECTION

In accordance with the requirements of 10CFR50, Appendix A, GDC 4, adequate missile protection is provided to ensure that those portions of the safety related structures, systems, or components whose failure would result in the failure of the integrity of the reactor coolant system pressure boundary, reduce to an unacceptable level the functioning of any plant feature required for a safe shutdown, or lead to unacceptable offsite radiological consequences, are designed and constructed so as not to fail or cause such a failure in the event of a postulated credible missile impact. The following sections provide the bases for the selection of the postulated missiles, protection requirements for external missiles, and details of the missile barrier design. Safety related systems or components are protected by locating them within missile-proof structures, by providing separation, or by providing missile shields or barriers. Nonsafety related structures, systems, and components are protected from internally generated missiles if their failure by postulated missile impact could prevent the required safety function of other safety related structures, systems, or components.

3.5.1 Missile Selection and Description

The following sources are considered for the generation of missiles:

- o Internally generated missiles:
  - Internally generated missiles outside containment.
  - Internally generated missiles inside containment.
- o Turbine missiles.
- o Externally generated missiles:
  - Missiles generated by natural phenomena.
  - Missiles generated by events near the site.

- Aircraft hazards.
- Gravity generated missiles.

The systems located both inside and outside of the containment have been examined to identify and classify potential missiles.

The basic approach is to ensure design adequacy against generation of missiles rather than to allow missile formation and design plant features to contain their effects. In those cases where missile formation does occur, plant features are designed to contain their effects.

3.5.1.1 Internally Generated Missiles (Outside Containment)

There are two general sources of postulated missiles outside the containment which are potentially generated as a result of plant operation:

- o Rotating component failures. (e.g., pump impeller, fan blade, turbine disk, motor, etc)

- o Pressurized component failures.

A tabulation of safety-related structures, systems, and components and their locations, seismic categories, and quality group classifications is given in Table 3.2-1. General arrangement and section detail drawings are located in Section 1.2 of RESAR-SP/90 PDA Module 3, "Introduction and Site".

3.5.1.1.1 Rotating Component Failure Missiles

Identification of missiles generated by postulated failure of rotating components, their sources and characteristics (i.e., size, mass, velocity, etc.), and location (including adjacent structures, casing thicknesses, etc) is made for a determination of the appropriate missile protection to be provided.

Missile selection is based on the following conditions:

A. All rotating components that are operated during normal operating plant conditions are considered to be potential missiles if the energy of the missile is sufficient to perforate the housing.

B. The energy in a rotating part associated with component failure is assumed to occur at 120 percent overspeed.

C. Components within one train of a redundant system are not protected from potential rotating missiles originating from the same train. Components within the other train are protected by complete separation and compartmentalization.

3.5.1.1.2 Pressurized Component Failure Missiles

Based on the design features noted below and review of the plant areas outside the containment containing pressurized components, there are no pressurized components whose failure will result in postulated missiles affecting the safety related systems, structures, and components required for safe shutdown of the reactor. The design features of the pressurized components and the basis for the postulated missiles selected are described below.

A. Pressurized components in systems which qualify as high-energy systems (as defined in Section 3.6) are evaluated as to their potential for becoming missiles.

B. Temperature detectors or other detectors installed in high-energy piping are evaluated as potential missiles if failure of a single circumferential weld could cause their ejection.

C. Where auxiliary fittings such as thermocouple wells, pressure gauges, vents, drains, and test connections are attached to piping or process equipment by threaded connections only, they are postulated as missiles. When such fittings are attached by welding and the completed joint has a greater design strength than the parent metal, they are not postulated as missiles.

D. Valves of American National Standards Institute (ANSI) rating 900 psig and above, constructed in accordance with Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, are pressure seal, bonnet-type valves. For pressure seal bonnet valves, valve bonnets are prevented from becoming missiles by the retaining ring, which would have to fail in shear, and by the yoke, which would capture the bonnet or reduce bonnet energy. Because of the highly conservative design of the retaining ring (safety factors in excess of eight may be used), bonnet ejection is highly improbable for these valves.

E. Most valves of ANSI rating 600 psig and below are valves with bolted bonnets. For the valve bonnet to be the source of a significant missile, rupture would take the form of a through-wall circumferential crack in the bonnet area. Such a crack would be detected as a leak long before it could propagate into a serious loss of fluid, or a missile generating failure. Valve bonnets are prevented from becoming missiles by limiting stresses in the bonnet-to-body bolting material by rules set forth in the ASME B&PV Code, Section III, and by designing flanges in accordance with applicable code requirements. Even if bolt failure were to occur, the likelihood of all bolts experiencing a simultaneous complete severance failure is very remote. The widespread use of valves with bolted bonnets and the low historical incidence of complete severance valve bonnet failures confirm that bolted bonnets need not be considered as credible missiles.

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

G. Nuts, bolts, nut and bolt combinations, and nut and stud combinations have only a small amount of stored energy and thus are not considered potential missiles.

H. Normally closed gate valves are not considered as potential missile sources, since the force of the fluid acts perpendicularly to the disc, stem, and operator.

I. Components within one train of a system containing redundant trains are not protected from potential pressurized missiles originating from the same train. Components within the other train are protected by complete separation and compartmentalization.

The conclusion, based on design features noted above, that valve bonnets are not credible missiles is also supported by industry experience.

3.5.1.2 Internally Generated Missiles (Inside Containment)

For systems located inside the reactor containment the basic approach is to assure design adequacy against generation of missiles, rather than allow missile formation and try to contain their effects. Failure of the reactor vessel, steam generators, pressurizer, reactor coolant pump castings, and piping leading to generation of missiles is not considered credible. This is because of the material characteristics, inspections, quality control during fabrication, erection and operation, conservative design, and prudent operation as applied to the particular component.

The two general sources of postulated missiles outside the containment (Subsection 3.5.1.1) also apply to inside the containment. A tabulation of missiles generated by postulated failure of rotating and pressure retaining components, their sources and characteristics (i.e., size, mass, velocity, etc), and location is made for the determination of appropriate missile protection to be provided. For the reactor coolant pressure boundary (RCPB), the selection of potential missiles is based on the application of single-failure criteria to the normal retention features of plant equipment for which there is a source of energy capable of creating a missile in the event of the postulated removal of the normal retention features. Where redundancy is provided by the normal retention features, such that sufficient retention capability remains to prevent creation of a missile in the event of a postulated failure of a single retention feature, no potential missile is postulated.

3.5.1.2.1 Control Rod Drive Mechanisms

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

A. Control rod drive mechanisms are shop hydrotested in excess of 150% of system design pressure.

B. Control rod drive mechanism housings are individually hydrotested to 125% of system design pressure after they are installed on the reactor vessel to the head adapters and are checked again during the hydrotest of the completed reactor coolant system.

C. Control rod drive mechanism housings are made of type 304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

D. Stress levels in the mechanisms are not affected by system transients at power or by thermal movement of the coolant loops.

However, for conservatism it is postulated that the top plug on the CRDM will become loose and will be forced upward by the water jet. CRDM missiles are contained by the integrated head missile shield.

3.5.1.2.2 Valves

Valves (as a pressure component) are evaluated to identify potential missiles. Design features described below, as well as in Subsection 3.5.1.1.2, preclude values from being considered as credible sources of missiles.

Valves with a nominal diameter larger than 2 inches are designed against bonnet-to-body connection failure and subsequent bonnet ejection by means of the following:

A. Compliance with the ASME Code, Section III.

B. Control of load during tightening of bonnet-to-body bolted connections.

Reactor coolant pressure retaining parts are constructed in accordance with the ASME B&PV Code, Section III, Class 1. The valves are hydrostatically tested in accordance with the ASME Code, Section III.

In the special case of those valves located on the top of the pressurizer, which extends above the operating deck, certain vertical missiles (although not considered credible) are postulated. Protection is provided by the concrete roof slab, which prevents potential damage to the containment, engineered safeguards components, and components located outside the pressurizer compartments.

3.5.1.2.3 Temperature and Pressure Sensors

The only credible source of jet-propelled missiles from the reactor coolant piping and piping systems connected to the reactor coolant system is that represented by the temperature and pressure sensor assemblies. The resistance temperature sensor assemblies can be of two types, i.e., with well and without well. Two rupture locations are postulated: one around the welding between the boss and the pipe wall, another at the welding (or thread) between the temperature element assembly and the boss for the without-well element and the welding (or thread) between the well and the boss for the with-well element.

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

The missile characteristics of these temperature and pressure element assemblies are not of concern from a containment penetration standpoint.

3.5 1.2.4 Other Missiles

The missile characteristics of the reactor coolant pump temperature sensor, the instrumentation well of the pressurizer, and the pressurizer heaters are also evaluated.

Pressurizer heaters are potential missiles; but inasmuch as they would be ejected in a downward direction, no damage to safety-related structures, systems, and components inside the containment would occur.

Consideration is also given to the reactor coolant pump flywheel as a potential missile source. For a discussion of the design, fabrication, inspection and material requirements placed on this component to assure its integrity, see Subsection 5.4.1.5 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System". In addition, provisions are made in the safety related circuitry to the pump motor to assure that specified overspeed limits are not exceeded even under faulted conditions; see Subsection 5.4.1.3 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System" for a discussion of the reactor coolant pump overspeed considerations. The degree of compliance of the pump flywheel with Regulatory Guide 1.14 is discussed in Section 5.4.1.5 and Section 1.8 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System".

The pressurizer relief tank rupture discs are designed such that their failure will not result in the formation of missiles. With rupture, the disc will split into quadrants that will be retained by the disc circumference. The tank is located low in the containment outside the secondary shield wall, and disc rupture will not cause failure to either the primary or secondary systems.

Based on the design features and the analysis presented in the preceding sections, it is concluded that because of compartmentalization, protective barriers, redundancy, and low kinetic energy associated with missiles, the intended safety function of the essential structures, systems, or components will not be impaired by any type of rotating or pressurized missile source.

3.5.1.3 Turbine Missiles

The turbine-generator and information related to turbine transient characteristics, potential missile generation and properties, placement and orientation, strike zones, missile probability analysis, overspeed protection, and turbine testing are not included in the scope of the Nuclear Power Block (NPB).

3.5.1.4 Missiles Generated by Natural Phenomena

The credible missiles at WAPWR created by natural phenomena are those generated by tornadoes. The design basis tornado missiles are based on the ANSI/ANS 2.3-1983, "Standard Design Missile Spectrum For Wind Velocity of 320 mph". The ANSI/ANS 2.3 standard is based on detailed analyses and evaluation of the data by experts leading to issue of the concensus standard in 1983. It represents more recent in-depth evaluation than was incorporated in Regulatory Guide 1.76 and the Standard Review Plan. The assessment of possible hazards due to these missiles is based on the guidelines of Regulatory Guide 1.117.

The methodology used to design the Category I structures to provide adequate protection for the safety-related equipment, system, and components is described in Section 3.5.3 of this module.

Safety related systems and components are protected by missile barriers. Where concrete exterior walls and roofs are used as barriers to offer missile protection, such walls have a 24-inch minimum thickness, while the roofs are at least 21 inches thick.

3.5.1.5 Missiles Generated by Events Near the Site

Although not part of the NPB, there are no credible site proximity missiles assumed by events near the WAPWR site.

3.5.1.6 Aircraft Hazards

Although not part of the NPB, no credible aircraft hazards to the WAPWR site are taken.

3.5.1.7 Gravity-Generated Missiles

The occurrence of falling objects as a result of seismic events is prevented by adequately supporting equipment in areas where the possibility of interaction exists. The postulated occurrence of falling objects as a result of the failure of a crane or hoist is discussed in RESAR-SP/90 PDA Module 13, "Auxiliary Systems."

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

3.5.2.1 General

The sources of missiles which, if generated, could affect the safety of the plant are considered in Subsection 3.5.1. Safety related structures, systems, and components are designed to withstand the impact of postulated missiles, are physically separated from the source of missiles, or are protected by a missile barrier. Included in this (as it relates to potential missiles) is the consideration of the guidelines of Regulatory Guide 1.13 to assure integrity of the spent fuel storages facility and fuel therein, and Regulatory Guide 1.117 for protection from the effects of the Design Basis tornado. Evaluation of the integrity of the facility ultimate heat sink and protection against turbine missiles is not part of the NPB.

3.5.2.2 Missile Barriers Within Containment

The secondary shield walls, the refueling canal walls, the various structural beams, and the operating floor act as missile barriers separating reactor coolant loops from other protected components and missile sources. These barriers also protect the reactor coolant pressure boundary (RCPB) in each loop from those identified missiles generated elsewhere in the containment building while protecting the RCPB in each loop from externally generated missiles. The feedwater system is routed so that it is not affected by potential missiles.

Except for short piping runs in the safety injection system (SIS) which must supply cooling water to the reactor coolant system after a loss of coolant accident, the emergency safety features are located outside the secondary shield. The SIS lines which penetrate the secondary shield do so in the vicinity of the loop segment to which they are attached.

A missile shield structure is provided over the control rod drive mechanisms (CRDMs) to block any identified missiles generated in that location. The design of the missile shield is discussed in Subsection 3.5.3. The control rod drives are protected from horizontal missiles by the refueling canal walls that extend vertically above the CRDMs. The head vent and letdown system piping is the only high-energy piping located close to the CRDMs. (No potential missile sources exist in the system.) A roof slab is provided to protect against identified missiles that originate in the region where the pressurizer extends above the operating floor.

Missile barriers are provided, as required, to prevent missiles generated by the failure of main steam or feedwater components inside the containment from causing loss of integrity of the containment, isolation system, or steam system associated with another steam generator, or from causing loss of function to other required systems or components inside the containment in accordance with the missile protection design criteria previously listed in Subsection 3.5.1.

3.5.2.3 Barriers for Missiles Generated Outside of Plant Structures

Protective structures, shields, and missile barriers are designed to provide protection against identified missiles generated outside these structures, shields, and missile barriers. The missile barriers are designed for the tornado and accident missiles described in Subsection 3.5.1 utilizing the approach stated in Subsection 3.5.3.

3.5.2.4 Missile Barriers Within Plant Structures Other Than Containment

Missile barriers are provided within plant structures outside the containment in conformance with the missile protection design criteria discussed in Section 3.5. For the pressurized and rotating component failure missiles that originate outside the containment, identified in Subsection 3.5.1, the following steps are taken to assure that the missile protection design criteria are met.

A. Missiles are categorized according to the system in which they originate.

B. The components that must be protected from a missile are identified in accordance with the missile protection design criteria given in Subsection 3.5.1.

3.5.3 Barrier Design Procedures

Missile barriers and protective structures are designed to withstand and absorb missile impact loads in order to prevent damage to safety-related components.

Formulae used for missile penetration calculations (for missiles other than turbine missile) into NPB steel or concrete barriers are:

<u>Concrete</u> (Modified Petry Formula)

$$D = K A_p \text{Log}_{10} \left(1 + \frac{V^2}{215,000}\right)$$

$$D' = D \left[1 + e^{-4(T/D-2)}\right]$$

where

D' = actual depth of penetration (ft)

D = depth of penetration for an infinite slab (ft)

T = thickness of the slab (ft)

Ap = sectional pressure $\frac{\text{missile weight}}{\text{impact area of missile}}$ (psf)

V = impact velocity (ft/sec)

K = experimentally obtained material coefficient for penetration

<u>Steel (Standard Formula)</u>

$$\frac{E}{D} = \frac{S}{46,500} \left(16,000\ T^2 + 1,500 \frac{W}{W_s}\ T\right)$$

where:

E = critical kinetic energy required for perforation (ft-lb)

D = missile diameter (in)

S = ultimate tensile strength of the target (steel plate) (psi)

T = target plate thickness (in)

W = length of a square side between rigid supports (in)

S_2 = length of a standard width (4 in)

The ultimate tensile strength is directly reduced by the amount of bilateral tension stress already in the target. The equation is good within the following ranges:

$0.1 < T/D < 0.8$,

$0.002 < T/L < 0.05$,

$10 < L/D < 50$,

$5 < W/D < 8$,

$8 < W/T < 100$,

$70 < V_c < 400$

Where:

L = missile length, in,

V_c = velocity, ft/sec, and

the missile is assumed to be cylindrical.

In using the Modified Petry and Stanford formulae for missile penetration it is assumed that the missile impacts normal to the plane of the wall on a minimum impact area and, in the case of reinforced concrete, does not strike the reinforcing. Due to the conservative nature of these assumptions, the minimum thickness required for missile shields will be taken as the thickness just perforated.

In applying the Stanford formula to the design of steel missile barriers, certain modifications to the formula are necessary to reflect the actual geometric and material properties of the missile and target under impact conditions. Test programs are continuing which, when completed, will define all the required modifications to the Stanford formula for the design of steel missiles barriers.

Secondary missiles caused by spalling of a concrete wall are of generally low energy and will therefore be neglected except where relatively fragile safety class equipment would be encountered. The thickness of a reinforced concrete wall which will just spall is calculated from the following formulae:

$$X = \frac{282\ W}{D^2\sqrt{f_c^{'}}}\ D^{0.215}\ \left(\frac{V}{1000}\right)^{1.5} + 0.5\ D$$

$$S = 2.28D + 1.13x$$

where:

S	= thickness to just scab (in)
x	= penetration in infinite concrete (in)
W	= weight of missile (lb)
D	= diameter of missile (in)
V	= striking velocity of missile (ft/sec.)
$f_c^{'}$	= compressive strength of concrete (psi)

The equations are stated to be good within the following ranges:

$1 \leq D \leq 16$

$0.4 \leq W \leq 2500$

$1500 \leq f_c^{'} \leq 8000$

$500 \leq V \leq 3000$

Structural members designed to resist missile impact will be designed for flexural, shear, and buckling effects using the equivalent static load obtained from the evaluation of structural response. Stress and strain limits for the equivalent static load will comply with the requirements of applicable codes or specifications except for the area local to the missile impact, where the stress and strain may exceed the allowables provided there will be no loss of function of any safety related system.

In general, Westinghouse-supplied equipment is not designed to withstand the impact of postulated missiles; therefore, the BOP designer considers the effects of postulated missiles and provides the necessary protection to safety related components as determined by the missile selection bases provided in Subsection 3.5.1.

The exception to this is the control rod drive mechanism (CRDM) missile shield, which is supplied by Westinghouse as part of the integrated head. A missile shield structure is provided over the CRDMs to block missiles that might be associated with a fracture of the pressure housing of any mechanism. This missile shield is a reinforced steel structure attached to the reactor vessel head and located above the CRDMs.

For the case of CRDM housing plug and drive shaft impact, which is the design case, it is assumed that the plug partially perforates the missile shield. The drive shaft then hits the plug and further penetrates the steel missile shield; the effective thickness of the steel missile shield is more than three times the combined penetration for the design case. The CRDM missile shield is also designed to withstand the dynamic impact loads due to the missile and the water jet.

3.5.4 Missile Protection Interface Requirements

The BOP applicant must consider the effects of postulated missiles and provide the necessary protection to safety related components as determined by the bases provided in Section 3.5 of this module. In general Westinghouse supplied equipment is not designed to withstand the impact of postulated missiles. Tables 3.5-1 through 3.5-4 list typical characteristics of missiles postulated inside the containment from equipment within the scope of the NPB.

All systems, equipment, and structures, identified as within the NPB scope in Table 3.2-1 and which are also required following a high-energy line break, must be evaluated for protection against missiles in Subsection 3.5 as well as those missiles identified by the BOP Applicant. However, design against postulated missiles is a function of plant layout, the missile generating source locations, the particular accident postulated at the time, etc., and only when all of this information is available can those safety related systems be evaluated for the degree of protection required.

Equipment within the NPB scope outside containment has been evaluated for potential missile sources. As a result of this review, the following information concerning potential missile sources and systems within the NPB scope which require protection from internally generated missiles outside containment is provided. The recommendations of standard ANSI N177, "Plant Design Against Missiles," have been followed.

Components within the NPB scope outside containment have been evaluated for potential missile sources. Valves in high pressure systems have been reviewed. As a result of this review, it is concluded that there are no credible sources of missiles associated with valves since there is no single failure associated with any potential valve parts that can result in the generation of a missile. Therefore, there are no postulated missiles associated with valves within the NPB scope outside containment.

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

The fabrication specifications of the MG set flywheels have control of material to meet ASTM-A533-7D, Grade B, Class I with inspections per MIL-I-45208A and flame cutting and machining operations governed to prevent flaws in the material. Nondestructive testing for nil-ductility (ASTM-E-208), charpy V-notch (ATM-A593), ultrasonic (ASTM-A578 and A579) and magnetic particles (ASTM Section III, NB2545) is performed on each flywheel material lot. In addition to these requirements stress calculations are performed consistent with guidelines of ASME Section III, Appendix A to show the combined primary stresses due to centrifugal forces and the shaft interference fit shall not exceed 1/3 of the yield strength at normal operating speeds (1800 rpm) and shall not exceed 2/3 of the yield strength at 25 percent overspeed. However, no overspeed is expected for the following reason: The flywheel dimensions are 32.26 inches in diameter x 4.76 inches wide and weighs approximately 1300 pounds. The flywheel mounted on the generator shaft which is directly coupled to the motor shaft, is driven by an 1800 rpm synchronous motor. The torque developed by the motor is insufficient for overspeed. Therefore, there are no credible missiles from the MG sets.

TABLE 3.5-1

SUMMARY OF TYPICAL CONTROL ROD DRIVE MECHANISM MISSILE ANALYSIS

Typical Examples of Postulated Missiles	Calculation Data: Missile Weight (lb)	Calculation Data: Impact Velocity (ft/sec)	Calculation Data: Kinetic Energy (ft-lb)	Penetration (in)	Assumptions
1. Mechanism Housing Plug	11	90	1,380	0.05	Plug becomes loose and is accelerated by the water jet.
2. Mechanism Housing Plug and Drive Shaft Impacting on same Missile Shield Spot	133	150	46,757	0.80	Drive shaft further pushes the plug into the shield.
3. Drive Shaft Latched to Mechanism	1,500	12.1	1,490	0.057	------

TABLE 3.5-2

VALVE - TYPICAL MISSILE CHARACTERISTICS

Missile Description	Weight (lb)	Flow Discharge Area (in^2)	Thrust Area (in^2)	To Impact Area (in^2)	Impact Area Ratio (psi)	Velocity (fps)
Safety Relief Valve Bonnet (3" x 6" or 6" x 6")	350	2.86	80	24	15.6	110
3 Inch Motor Operated Isolation Valve Bonnet (plus motor and stem) (3")	400	5.5	113	28	14.1	135
2 Inch Air Operated Relief Valve Bonnet (plus stem)	75	1.8	20	20	3.75	115
3 Inch Air Operated Spray Valve Bonnet (plus stem)	120	5.5	50	50	2.4	190
4 Inch Air Operated Spray Valve	200	9.3	50	50	4	190

TABLE 3.5-3

PIPING TEMPERATURE ELEMENT ASSEMBLY - TYPICAL MISSILE CHARACTERISTICS

1. For a tear around the weld between the boss and the pipe:

Characteristics	"without well"	"with well"
Flow Discharge Area	0.11 in^2	0.60 in^2
Thrust Area	7.1 in^2	9.6 in^2
Missile Weight	11.0 lb	15.2 lb
Area of Impact	3.14 in^2	3.14 in^2
$\left[\frac{\text{Missile Weight}}{\text{Impact Area}}\right]$	3.5 psi	4.84 psi
Velocity	20 ft/sec	120 ft/sec

2. For a tear at the junction between the temperature element assembly and the boss for the "without well" element and at the junction between the boss and the well for the "with well" element.

Characteristics	"without well"	"with well"
Flow Discharge Area	0.11 in^2	0.60 in^2
Thrust Area	3.14 in^2	3.14 in^2
Missile Weight	4.0 lb	6.1 lb
Area of Impact	3.14 in^2	3.14 in^2
$\left[\frac{\text{Missile Weight}}{\text{Impact Area}}\right]$	1.27 psi	1.94 psi
Velocity	20 ft/sec	120 ft/sec

TABLE 3.5-4

TYPICAL CHARACTERISTICS OF OTHER MISSILES POSTULATED WITHIN REACTOR CONTAINMENT

	Reactor Coolant Pump Temperature Element	Instrument Wall of Pressurizer	Pressurizer Heaters
Weight	0.25 lb	55 lb	15 lb
Discharge Area	0.50 in^2	0.442 in^2	0.80 in^2
Thrust Area	0.50 in^2	1.35 in^2	2.4 in^2
Impact Area	0.50 in^2	1.35 in^2	2.4 in^2
$[\frac{\text{Missile Weight}}{\text{Impact Area}}]$	0.5 psi	4.1 psi	6.25 psi
Velocity	260 ft/sec	100 ft/sec	55 ft/sec

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

In the event of the high- or moderate-energy pipe failure within the plant, adequate protection is provided to ensure that essential structures, systems, or components are not impacted by the effects of postulated piping failure. Essential systems and components are those required to shut down the reactor and mitigate the consequences of the postulated piping failure.

The following sections provide the bases for selection of the pipe failure locations, and the determination of the resultant effects.

3.6.1 Postulated Piping Failures in Fluid Systems Inside and Outside Containment

The criteria and methods used to address postulated piping failures are described below. Table 3.6-1 provides a matrix of plant systems that indicates their classification: high-energy, and/or essential. Selection of pipe failure locations and evaluation of the consequences on nearby essential systems, components, and structures are presented in Subsection 3.6.2.

3.6.1.1 Design Bases

The following design bases relate to the evaluation of the effects of the pipe failures determined in Subsection 3.6.2:

A. The selection of the failure type is based on whether the system is high or moderate-energy during normal operating conditions of the system.

   High-energy piping includes those systems or portions of systems in which the maximum normal operating temperature exceeds 200°F or the maximum normal operating pressure exceeds 275 psig.

Piping systems or portions of systems pressurized above atmospheric pressure during normal plant conditions and not identified as high-energy are considered moderate energy.

Piping systems that exceed 200°F or 275 psig for about 2 percent or less of the time the system is in operation or that experience high-energy pressures or temperatures for less than 1 percent of the plant operation time are considered moderate-energy.

B. The following assumptions are used to determine the thermodynamic state in the piping system for the calculation of fluid reaction forces:

For those portions of piping systems normally pressurized during operation at power, the thermodynamic state in the pipe and associated reservoirs are those of full (100-percent) power operation.

C. Moderate-energy pipe cracks are evaluated for spray wetting, flooding, and other environmental effects.

D. Where postulated, each longitudinal or circumferential break in high-energy fluid system piping or leakage crack in moderate-energy fluid system piping is considered separately as a single initial event occurring during normal plant conditions.

E. Offsite power is assumed to be unavailable if a trip of the turbine-generator system or reactor protection system is a direct consequence of the postulated piping failure.

F. A single active component failure is assumed in systems used to mitigate the consequences of the postulated piping failure or to shut down the reactor, except as noted in paragraph G below. The single active component failure is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure, such as unit trip and loss of offsite power.

G. When the postulated piping failure occurs in one of two or more redundant trains of a dual-purpose, moderate-energy essential system, single failures of components in other trains (and associated supporting trains) are not assumed; this is because the system is designed to Seismic Category I standards, powered from both offsite and onsite sources, and constructed, operated, and inspected to quality assurance, testing, and inservice inspection standards appropriate for nuclear safety systems.

H. All available systems, including those actuated by operator actions, are employed to mitigate the consequences of a postulated piping failure to the extent clarified in the following paragraphs:

1. In determining the availability of the systems, 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 determined on the basis of ample time and adequate access to equipment being available for the proposed actions. Although a postulated high/moderate-energy line failure outside the containment may ultimately require a cold shutdown, operation at hot standby is allowed in order for plant personnel to assess the situation and make repairs.

2. The use of non-Seismic Category I piping in mitigating the consequence of postulated piping failure outside the containment is clarified in the following paragraphs:

   a. For non-Seismic Category I piping which is not seismically supported, it is assumed that a safe shutdown earthquake could be the cause of the failure. Therefore, only Seismic Category I equipment and seismically supported non-category I equipment can be used to mitigate the consequences of the failure and bring the plant to a safe shutdown.

b. Category I and seismically supported non-Category I piping systems located outside the containment are assumed to fail nonmechanistically (i.e., failure is produced by some mechanism other than an earthquake) for the purpose of pipe break hazard analysis. Therefore, non-Category I equipment can be used to bring the plant to a safe shutdown following a postulated pipe break event, subject to the power being available to operate such equipment and provided that the radiological consequences are insignificant in comparison to 10 CFR 100 dose guidelines.

I. A whipping pipe is not considered capable of rupturing impacted pipes of equal or greater nominal pipe diameter and equal or greater wall thickness. This is based on the assumption that only piping is determined to do the impacting. A whipping pipe is considered capable of developing a through-wall leakage crack in a pipe of larger nominal pipe size with thinner wall thickness, assuming that only piping is determined to do the impacting. The above criterion is not utilized where the potential exists for valves or other components in the whipping pipe to impact the targets, since these are treated on a case-by-case basis.

J. Pipe whip is assumed to occur in the plane defined by the piping geometry and to cause movement in the direction of the jet reaction.

If unrestrained, a whipping pipe having a constant energy source sufficient to form a plastic hinge is considered to form a plastic hinge and rotate about the nearest rigid pipe whip restraint, anchor, or wall penetration capable of resisting the pipe whip loads. If the direction of the initial pipe movement caused by the thrust force is such that the whipping pipe impacts a flat surface normal to its direction of travel, it is assumed that the pipe comes to rest against that surface, with no pipe whip in other directions.

In general, whipping ends from a pipe break are restrained so that plastic hinge formation is not allowed to occur. Where a plastic hinge could be formed, the effects are evaluated. Pipe whip restraints are provided wherever postulated pipe breaks could impair the ability of any essential system or component to perform its intended safety functions.

K. The calculation of thrust and jet impingement forces considers any line restrictions (e.g., flow limiter) between the pressure source and break location and the absence of energy reservoirs, as applicable.

L. Pipe breaks are not postulated to occur in pump and valve bodies since the wall thickness exceeds that of connecting pipe.

M. Pipe breaks are not postulated to occur in systems for which postulated through-wall cracks have been shown to be stable for worst case loadings (See Subsection 3.6.2.1.1E for a listing of these systems). Leak detection systems are provided which are capable of detecting the leakage from a postulated through-wall crack.

3.6.1.2 Description

Will be provided in RESAR-SP/90 FDA version.

3.6.1.3 Safety Evaluation

Will be provided in RESAR-SP/90 FDA version.

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

This subsection describes the design bases for locating postulated breaks and cracks in high and moderate-energy piping systems inside and outside of the containment, the procedures used to define the jet thrust reaction at the break location, the procedures used to define the jet impingement loading on adjacent essential structures, systems, or components, pipe whip restraint design, and the protective assembly design.

3.6.2.1 Criteria Used to Define High/Moderate-Energy Break/Crack Locations and Configurations

A postulated high-energy pipe break is defined as a sudden, gross failure of the pressure boundary of a pipe either in the form of a complete circumferential severance (i.e., a guillotine break) or as a sudden longitudinal, uncontrolled crack. For moderate-energy fluid systems, pipe failures are confined to postulation of controlled cracks in piping. The effects of these cracks on the safety-related equipment are analyzed for flooding and wetting only. These cracks do not result in jet impingement or whipping of the cracked piping.

3.6.2.1.1 High-Energy Break Locations

With the exception of those portions of the piping identified in Subsections 3.6.2.1.1.D or E, breaks are postulated in high-energy piping at the following locations:

A. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B and PV) Code, Section III, Division 1 - Class 1 Piping

   1. In the reactor coolant system primary loops, there are no postulated break locations based on the criteria of Subsection 3.6.2.1.1.E.

   2. Pipe breaks are postulated to occur at the following locations in Class 1 piping runs or branch runs outside the primary RCLs as follows:

      a. At terminal ends of the piping, including:

         (1) Piping connected to structures, components, or anchors that act as essentially rigid restraints to piping translation and rotational motion due to static or dynamic loading.

(2) Branch intersection points are considered a terminal end for the branch line unless the following are met: the branch and the main piping systems are modeled in the same static, dynamic, and thermal analyses, and the branch and main run are of comparable size and fixity, i.e., the nominal size of the branch is at least one-half of that of the main.

b. At all intermediate locations where the following conditions are satisified:

(1) Any intermediate locations where the maximum stress range as calculated by equation (10) and either (12) or (13) exceeds 2.8 S_m (where S_m is the design stress intensity) as described in paragraph NB-3653 of the ASME B and PV Code, Section III.

(2) Any intermediate locations where the cumulative usage factor exceeds 0.25.

B. ASME B and PV Code, Section III - Class 2 and 3 Piping Systems

1. Pipe breaks are postulated to occur at terminal ends.

2. Pipe breaks are postulated at intermediate locations between terminal ends where the maximum stress value, as calculated by the sum of equations (9) and (10) in subarticle NC-3652 of the ASME B and PV Code, Section III, Reference 3, considering normal and upset plant conditions (i.e., sustained loads, occasional loads, thermal expansion, and an operating basis earthquake (OBE) event) exceeds 0.9 (1.8 S_h + S_A).

S_h and S_A are the allowable stress at maximum hot temperature and allowable stress range for thermal expansion, respectively, for Class 2 and 3 piping, as defined in subarticle NC-3600 of the ASME B and PV Code, Section III.

C. Nonnuclear Piping (i.e., not ASME Section III Class 1, 2, or 3)

Breaks in nonnuclear piping are postulated at the following locations in each run:

1. At the locations specified for ASME Section III, Reference 1, Class 2 and 3 piping (refer to Subsection 3.6.2.1.1.B), if the nonnuclear piping is analyzed and supported to withstand full safe shutdown earthquake loadings.

2. In the absence of stress analysis, breaks in nonnuclear piping are postulated at the following locations in each run or branch run:

   a. Terminal ends.

   b. Each intermediate fitting, e.g., short and long radius elbows, tees, and reducers; welded attachments; and valves.

D. High-Energy Piping in Containment Penetration Areas

Breaks are not postulated in the portions of Class 2 piping between the containment penetration flued-head and five-way restraints (i.e., break exclusion zone) provided subject piping meets the following provisions:

1. Stresses do not exceed those specified in Subsection 3.6.2.1.1.B.

2. The maximum stress in this piping as calculated by equation (9), per paragraph NC-3652 of ASME Section III, when subjected to the combined loadings of internal pressure, deadweight, and pipe rupture outside the protective restraints, does not exceed 1.8 S_h.

3. The number of circumferential and longitudinal piping welds and branch connections is minimized.

Areas of system piping where no breaks are postulated are as follows:

a. The main steam piping, from the containment penetration flued-head outboard weld, to the upstream weld of the five-way restraint, which is downstream of the main steam isolation valves, including the main steam safety valves and branch piping to the main steam safety valves.

b. The main feedwater piping from the containment penetration to the five-way restraint which is upstream of the isolation valve.

When required for isolation valve operability, structural integrity, or containment integrity, five-way restraints capable of resisting torsional and bending moments produced by a postulated pipe break, either upstream or downstream of the piping and valves which form the containment isolation boundary, are located reasonably close to the isolation valves or penetration.

The five-way restraints do not prevent the access required to conduct inservice inspection examinations specified in Section XI of the ASME Code. Inservice examinations completed during each inspection interval provide examination of circumferential and longitudinal pipe welds within the boundary of these portions of piping during each inspection interval.

Welded attachments to these portions of piping for pipe supports or other purposes are avoided. Where welded attachments are necessary, detailed stress analyses are performed to demonstrate compliance with the limits of Subsection 3.6.2.1.1.

The five-way restraints outside the containment on the main steam and main feedwater lines are located as close as possible to the containment to accommodate the design for the reactor external building and main steam tunnel and still minimize stresses.

For evaluation of environmental effects (excluding jet impingement) longitudinal breaks, with break flow areas of 1.0 square feet, are postulated in the main steam and feedwater piping. Locations which have the greatest effect on essential equipment are chosen.

E. Piping Within Mechanistic Pipe Break Criteria

The criteria below are used to verify that there are no pipe break locations in lines greater than 6 inches nominal diameter in the following high energy systems:

Reactor coolant
Emergency core cooling
Chemical and volume control
Main steam
Main feedwater
Steam generator blowdown
Diesel generator and related systems

The mechanistic pipe break approach is used instead of hypothetically located pipe ruptures and eliminates the structural analysis associated with these ruptures. Application of this approach is applied to high-energy piping provided:

a. Operating experience, tests, or analyses have indicated no particular susceptibility to failure from effects of intergranular stress corrosion cracking, water hammer, or thermal fatigue.

b. Supports of heavy interconnected components (such as reactor vessel, steam generator, and main reactor coolant pump in the reactor coolant system) are designed to withstand normal operation and SSE loads, and loads resulting from any postulated pipe rupture.

Dynamic effects associated with hypothetical full flow area circumferential or longitudinal breaks in the piping need not be considered when application of the mechanistic pipe break approach is technically justifiable in accordance with the evaluation criteria described below. The specific dynamic effects excluded are:

a. Pipe whip and reaction forces.
b. Jet impingement loads.[a]
c. Subcompartment pressurization such as reactor cavity asymmetric pressurization transients.
d. Break associated transient loads in unbroken portions of the system such as loads on the reactor internals or steam generator internals and pump overspeed.

The following requirements apply to high-energy piping inside or outside containment:

a. For purposes of specifying design criteria for emergency core cooling, containment systems, other non-structural engineered safety features, and for the evaluation of environmental effects, loss of coolant (even in the piping with applicability of the mechanistic pipe break approach) is assumed through an opening equivalent to twice the flow area of the largest diameter pipe in the system, or that pipe which will result in the most limiting accident conditions.

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a. However, environmental effects, wetting and flooding of surrounding equipment, and spaces due to leakage must be considered.

b. Except as required by (a) above, high-energy piping may be treated with potential of through-wall crack leakage rates equal to: 1) the maximum allowable unidentified leakage conditions associated with the piping, or; 2) the leakage for a rectangular crack having a one-half wall thickness width and a one-half pipe diameter length.

The following information is developed in the RESAR-SP/90 FDA for each line for which the mechanistic pipe break approach is applied:

a. A discussion to support a conclusion that the line is very unlikely to experience stress corrosion cracking, or extreme repetitive loads, or excessive loads such as might occur from thermal or mechanical low and high cycle fatigue or a water hammer.

b. Identification of types and specifications of all concerned materials; all base metal, forgings and weldments, and safe-ends will be included. The materials properties data and information used in the analysis will be provided, and the sources of all data reported.

c. Specification of the type and magnitude of the loads applied (forces, bending and torsional moments), their source(s) and method of combination. Identification of the location(s) at which the minimum margin (e.g., stress-to-strength ratio) occurs for base materials and weldments and safe-ends. For geometrically complex lines or systems, it may be necessary to analyze several locations to assure that the more limiting locations are identified.

Step-Wise Analysis Criteria

The following analytical steps, illustrated in Figure 3.6-1, assume that circumferentially oriented postulated cracks are limiting. If this is not the case, then the analysis described in (a) through (c) below will also include the postulation of axial cracks and/or elbow cracks. If applied moments (including SSE) are quite low and applied maximum axial forces dominate, relatively long part-through-wall cracks are analyzed to demonstrate that they are stable.

a. Postulated Fabrication Flaw

At the location or locations of (c) above, postulate a fabrication flaw that may be missed during fabrication and preservice inspections or would be permitted by code, whichever is larger. Demonstrate by fatigue analysis that the crack will not grow through the wall or extend significantly in length during plant design life.

b. Postulated Leakage Crack

Even though (a) above demonstrates that a leaking pipe is unlikely, a through-wall crack at the selected location is postulated. The size of the postulated crack should be large enough so that the leakage is assured of detection with adequate margin using 2 times the minimum installed leak detection capability when the pipe(s) is (are) subjected to normal operational loads. If auxiliary leak detection systems are relied on, they will be described.

c. Stability and Critical Crack Sizes

Demonstrate crack size margin by showing that 2 times the postulated leakage crack as defined in (b) above is less than the critical crack size using normal plus SSE loads. In some cases, a limit load analysis may suffice for this purpose, however, an elastic-plastic fracture mechanics analysis may be used when applicable.

3.6.2.1.2 Types of Breaks/Cracks Postulated

3.6.2.1.2.1 ASME Section III Piping Other than RCL Piping - High-Energy

The following types of breaks are postulated to occur at the locations determined in accordance with Subsection 3.6.2.1.1 - A, B and C.

A. In piping of 4 inches nominal diameter or greater, both circumferential and longitudinal breaks are postulated at each selected break location unless eliminated by comparison of longitudinal and axial stresses with the maximum stress as follows:

   1. If the maximum stress range exceeds the limits specified in Subsection 3.6.2.1.1.A.2.b or 3.6.2.1.1.B.2 but the circumferential stress range is at least 1.5 times the axial stress range, only a circumferential break is postulated.

   2. If the maximum stress range exceeds the limits specified in Subsections 3.6.2.1.1.A.2.b or 3.6.2.1.1.B.2 but the axial stress is at least 1.5 times the circumferential stress range, only a longitudinal break is postulated.

   Longitudinal breaks, however, are not postulated at terminal ends.

B. In piping of nominal diameter greater than 1 inch but less than 4 inches, only circumferential breaks are postulated at each selected break location.

C. No breaks are postulated for piping of nominal diameter 1 inch or less.

3.6.2.1.2.2 Nonnuclear Piping - High-Energy

The types of breaks for high-energy nonnuclear piping are postulated as discussed in Subsection 3.6.2.1.2.1; the corresponding break locations are determined in accordance with Subsection 3.6.2.1.1.C.

3.6.2.1.2.3 ASME Section III and Nonnuclear Piping - Moderate-Energy

Through-wall leakage cracks are postulated in moderate-energy piping including branch runs larger than 1 inch nominal diameter as clarified below:

A. Through-wall leakage cracks are not postulated in those portions of piping between containment isolation valves, provided they meet the requirements of ASME Code, Section III, subarticle NE-1120, and are designed so that the maximum stress range does not exceed 0.45 (1.8 $S_h + S_A$).

B. Through-wall leakage cracks are not to be postulated in moderate-energy fluid system piping located in an area where a break in the high-energy fluid system is postulated, provided that such cracks do not result in environmental conditions more limiting than the high-energy pipe break.

C. Through-wall leakage cracks are to be postulated in:

(1) ASME, B and PV Code, Section III, Division 1 - Class 1 piping where the maximum stress range in the piping is greater than 1.4 S_m.

(2) ASME, B and PV Code, Section III, Division 1 - Class 2 or 3 piping and seismically supported nonnuclear class piping at locations where the maximum stress range in the piping is greater than 0.45 (1.8 $S_h + S_A$).

To simplify analysis, cracks may be postulated to occur everywhere in moderate-energy piping regardless of the stress analysis results to determine the maximum damage from fluid spraying and flooding, with the consequent hazards or environmental conditions. Flooding effects are determined on the basis of 30 minutes operator time required to effect corrective actions.

3.6.2.1.3 Break/Crack Configuration

3.6.2.1.3.1 High-Energy Break Configuration

Following a circumferential break, the two ends of the broken pipe are assumed to move clear of each other unless physically limited by piping restraints, structural members, or piping stiffness. The effective cross-sectional (inside diameter) flow area of the pipe is used in the jet discharge evaluation. Movement is assumed to be in the direction of the jet reaction initially with the total path controlled by the piping geometry.

The orientation of a longitudinal break, except when otherwise justified by a detailed stress analysis, is assumed to be at opposing points on a line perpendicular to the plane of a fitting for a nonaxisymmetric fitting and anywhere around the circumference of the fitting for axisymmetric fittings. The flow area of such a break is equal to the cross-sectional flow area of the pipe. Both circumferential and longitudinal breaks are postulated to occur, but not concurrently, in all high-energy piping systems at the locations specified in Subsection 3.6.2.1.1, except as follows:

a. Circumferential breaks are not postulated in piping runs of 1-inch nominal diameter or less.

b. Longitudinal breaks are not postulated in piping runs of a nominal diameter less than four inches.

c. Longitudinal breaks are not postulated at intermediate locations in piping runs where the stress and cumulative usage factor limits for postulating intermediate rupture locations as specified in Subsection 3.6.2.1.1 for Class 1 piping and for Class 2 and 3 piping are not exceeded.

d. Longitudinal breaks are not postulated at terminal ends.

e. Only one type of break is postulated at locations where, from a detailed stress analysis such as a finite element analysis, the state of stress can 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.

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

   1. Without the benefit of a detailed stress analysis, such as a finite element analysis, circumferential breaks are postulated to occur individually at each pipe-to-fitting weld, and longitudinal breaks postulated to occur individually (except in piping with a nominal diameter less than four inches) on each side of the fitting at its center and oriented perpendicular to the plane of the fitting, or

   2. Alternatively, if a detailed stress analysis or test is performed, the results may be used to predict the most probable rupture location(s) and type of break.

g. Where the postulated break location is at a branch run connection, a circumferential break is postulated at the branch run pipe-to-fitting weld.

h. Where the postulated break location is at a welded attachment (lugs, stanchions, etc.) a circumferential break is postulated at the centerline of the welded attachment.

i. Where the postulated break location is at a reducer, circumferential breaks are postulated at each pipe-to-fitting weld. Longitudinal breaks are oriented to produce out-of-plane bending of the piping configuration on both sides of the reducer at each pipe-to-fitting weld.

3.6.2.1.3.2 Moderate-Energy Crack Configuration

Moderate-energy crack openings are assumed to be a circular orifice with cross-sectional flow area equal to that of a rectangle one-half the pipe inside diameter in length and one-half wall thickness in width.

3.6.2.2 Analytical Methods to Define Forcing Functions and Response Models

3.6.2.2.1 Forcing Functions for Jet Thrust

To determine the forcing function for means identified by Subsection 3.6.2.1.1-A, B and C, the fluid conditions at the upstream source and at the break exit dictate the analytical approach and approximations that are used. For most applications, one of the following situations exists:

- o Superheated or saturated steam.
- o Saturated or subcooled water.
- o Cold water (nonflashing).

Analytical methods for calculation of jet thrust for the above-described situations are discussed in Reference 5.

3.6.2.2.1.1 Time Functions of Jet Thrust Force on Intact Reactor Coolant Loop (RCL) Piping

To determine the thrust and reactive force loads to be applied to the RCL during the postulated loss-of-coolant accident (LOCA) in Subsection 3.6.2.1.1-A, B and C, it is necessary to have a detailed description of the hydraulic transient. Hydraulic forcing functions are calculated for the intact RCLs as a result of a postulated LOCA in branch runs connecting to the primary RCL. These forces result from the transient flow and pressure histories in the reactor coolant system (RCS). The calculation is performed in two steps. The first step is to calculate the transient pressure, mass flowrates, and thermodynamic properties as a function of time. The second step uses the results obtained from the hydraulic analysis, along with input of areas and direction coordinates, and calculates the time-history of forces at appropriate locations (e.g., elbows) in the RCLs.

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

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

The MULTIFLEX computer code calculates the hydraulic transients within the entire primary coolant system. This hydraulic program considers a coupled, fluid-structure interaction by accounting for the deflection of the core support barrel. The depressurization of the system is calculated using the method of characteristics applicable to transient flow of a homogeneous fluid in thermal equilibrium.

The ability to treat multiple flow branches and a large number of mesh points gives the MULTIFLEX code the flexibility required to represent the various flow passages within the primary RCS. The system geometry is represented by a network of one-dimensional flow passages.

The THRUST computer program was developed to compute the transient (blowdown) hydraulic forces resulting from a LOCA. The THRUST code calculates forces exactly the same way as the STHRUST code which is described in Reference 3.

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

$$F = 144A\ \left((P - 14.7) + \frac{\dot{m}^2}{\rho g\ A_m^2 \times 144}\right)$$

The symbols and units are as follows:

F = Force (lb_f)

A = Aperture area (ft^2)

P = System pressure (psia)

$\dot{m}$ = Mass flowrate (lbm/s)

ρ = Density (lbm/ft^3)

g = Gravitational constant = 32.174 ft-lbm/lb $-s^2$

A_m = Mass flow area (ft^2)

In the model to compute forcing functions, the RCL system is represented by a model similar to that employed in the blowdown analysis. The entire loop layout is represented in a global coordinate system. Each node is fully described by:

A. Blowdown hydraulic information.

B. The orientation of the streamlines of the force nodes in the system, which includes flow areas, and projection coefficients along the three axes of the global coordinate system.

Each node is modeled as a separate control volume with one or two flow apertures associated with it. Two apertures are used to simulate a change in flow direction and area. Each force is divided into its x, y, and z components using the projection coefficients. The force components are then summed over the total number of apertures in any one node to give a total x force, a total y force, and a total z force. These thrust forces serve as input to the piping/restraint dynamic analysis.

3.6.2.2.2 Dynamic Analysis of the Reactor Coolant Loop Piping and Equipment Supports

The dynamic analysis of the RCL for LOCA loadings is described in Section 3.9.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.3.1 Dynamic Analysis Methods to Verify Integrity and Operability for Other than RCL

The analytical methods of Reference 4 are used to determine the jet impingement effects and loading effects applicable to components and systems resulting from postulated pipe breaks.

3.6.2.3.2 Dynamic Analysis Methods to Verify Integrity and Operability for the RCL

3.6.2.3.2.1 General

A LOCA is assumed to occur for a postulated branch line break in Subsection 3.6.2.1.1-A down to the restraint of the second normally open automatic isolation valve (Case II, Figure 3.6-2) on outgoing lines[a] and down to and including the second check valve (Case III, Figure 3.6-2) on incoming lines

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a. It is assumed that the motion of the unsupported line containing the isolation valves can cause failure of the operators of both valves to function.

normally with flow. A pipe break beyond the restraint or second check valve does not result in an uncontrolled loss of reactor coolant if either of the two valves in the line closes.

Accordingly, both of the automatic isolation valves are suitably protected and restrained as close to the valves as possible so that a pipe break beyond the restraint does not jeopardize the integrity and operability of the valves. Further, periodic testing capability of the valves to perform their intended function is essential. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Cases I and IV, Figure 3.6-2), a LOCA is assumed to occur for pipe breaks on the reactor side of the valve. Branch lines connected to the RCL are defined as large strictly for the purpose of pipe break criteria if they have an inside diameter greater than 4 inches. Rupture of these lines results in a rapid blowdown from the RCL, and protection is basically provided by the accumulators and the low-head safety injection pumps (residual heat removal pumps).

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

Engineered safety features are provided for core cooling and boration, pressure reduction, and activity confinement in the event of a LOCA or steam or feedwater line break accident to ensure that the public is protected in accordance with 10 CFR 100 guidelines. These safety systems are described in Subsection 6.2.1.2 of RESAR-SP/90 PDA Module 10, "Containment Systems."

To assure the continued integrity of the essential components and the engineered safety systems, consideration is given to the consequential effects of the pipe break itself to the extent that:

A. The minimum performance capabilities of the engineered safety systems are not reduced below that required to protect against the postulated break.

B. The containment leak tightness is not decreased below the design value if the break leads to a LOCA.[a]

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

   1. A pipe break which is not a LOCA will not cause a LOCA or steam or feedwater line break. However, a break which is not a LOCA is permitted to propagate to a single 0.375 inch diameter primary side line provided that line is not part of a post accident monitoring system.

   2. An RCS pipe break will not cause a steam or feedwater system pipe break, and vice versa.

3.6.2.3.2.2 Large RCS Piping

Large branch line piping, as defined in Subsection 3.6.2.3.2.1, is restrained to meet the following criteria in addition to items A through C of Subsection 3.6.2.3.2.1 for a pipe break resulting in a LOCA:

A. Propagation of the break postulated in accordance with Subsection 3.6.2.1.1-A to the unaffected loops is prevented to ensure the delivery capacity of the accumulators and low head pumps.

B. Propagation of the break postulated in accordance with Subsection 3.6.2.1.1-A in the affected loop is permitted to occur but does not exceed 20 percent of the flow area of the line which initially ruptured. This criterion is voluntarily applied so as not to increase substantially the severity of the LOCA.

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a. The containment is here defined as the containment structure and penetrations, the steam generator shell, the steam generator steam side instrumentation connections, the steam, feedwater, blowdown, and steam generator drain pipes within the containment structure.

3.6.2.3.2.3 Small Branch Lines

Should one of the small pressurized lines, as defined in Subsection 3.6.2.3.2.1, fail and result in a LOCA, the piping is restrained or arranged to meet the following criteria in addition to items A through C of Subsection 3.6.2.3.2.1:

A. Break propagation is limited to the affected leg, i.e., propagation to the other leg of the affected loop and to the other loops is prevented. However, a break is permitted to propagate to a single 0.375 inch diameter line attached to another leg of the affected loop provided that line is not part of a post accident monitoring system. Damage to the high-head safety injection lines connected to the other leg of the affected loop or to the other loops is prevented.

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

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

3.6.2.3.3 Types of Pipe Whip Restraints

3.6.2.3.3.1 Pipe Whip Restraints

To satisfy varying requirements of available space, permissible pipe deflection, and equipment operability, the restraints are designed as a combination of an energy-absorbing element and a restraining structure suitable for the geometry required to pass the restraint load from the whipping pipe to the main building structure.

The restraint structure is typically a structural steel frame or truss and the energy-absorbing element is usually either stainless steel U-bars or energy-absorbing material as described below:

A. Stainless Steel U-Bar

This type consists of one or more U-shaped, upset-threaded rods of stainless steel looped around the pipe but not in contact with the pipe to allow unimpeded pipe motion during seismic and thermal movement of the pipe. At rupture, the pipe moves against the U-bars, which absorb the kinetic energy of pipe motion by yielding plastically. A typical example of a U-bar restraint is shown in Figure 3.6-3.

B. Energy Absorbing Material

This type of restraint consists of a crushable, stainless steel, internally honeycomb-shaped element designed to yield plastically under impact of the whipping pipe. A design hot position gap is provided between the pipe and the energy-absorbing material to allow unimpeded pipe motion during seismic and thermal pipe movements. A typical example of an energy-absorbing material restraint is shown in Figure 3.6-4.

3.6.2.3.4 Analytical Methods

3.6.2.3.4.1 Pipe Whip Restraints

A. Location of Restraints

1. For purposes of determining pipe hinge length and thus locating the pipe whip restraints, the plastic moment of the pipe is determined in the following manner:

$$M_p = 1.1\ z_p S_y$$

where:

z_p = Plastic section modulus of pipe

S_y = Yield stress at pipe operating temperature.

1.1 = 10-percent factor to account for strain hardening.

Pipe whip restraints are located as close to the axis of the reaction thrust force break as practicable. Pipe whip restraints are generally located so that a plastic hinge does not form in the pipe. If, due to physical limitations, pipe whip restraints are located so that a plastic hinge can form, the consequences of the whipping pipe and the jet impingement effect are further investigated. Lateral guides are provided where necessary to predict and control pipe motion.

2. Generally, restraints are designed and located with sufficient clearances between the pipe and the restraint such that they do not interact and cause additional piping stresses. A design hot position gap is provided that will allow maximum predicted thermal, seismic, and seismic anchor movement displacements to occur without interaction.

   Exceptions to this general criterion may occur when a pipe support and restraint are incorporated into the same structural steel frame, or when a zero design gap is required. In these cases the restraint is included in the piping analysis, if required.

3. In general, the restraints do not prevent the access required to conduct inservice inspection examination of piping welds. When the location of the restraint makes the piping welds inaccessible for inservice inspection, a portion of the restraint is made removable to provide accessibility.

B. Analysis and Design

Analysis and design of pipe whip restraints for postulated pipe break effects are in accordance with Reference 5. Specifically, the following criteria are adopted in analysis and design:

1. Pipe whip restraints are designed based on energy absorption principles by considering the elastic-plastic, strain-hardening behavior of the materials used.

2. A rebound factor of 1.1 is applied to the jet thrust force.

3. Except in cases where calculations are performed to verify that a plastic hinge is formed, the energy absorbed by the ruptured pipe is conservatively assumed to be zero; i.e., the thrust force developed goes directly into moving the broken pipe and is not reduced by the force required to bend the pipe.

4. In elastic-plastic design, limits for strains are as follows:

   ϵ = Allowable strain used in design.

   a. Stainless Steel U-Bars

   $\epsilon = 0.5\epsilon_u$

   where:

   ϵ_u = ultimate uniform strain of stainless steel (strain at ultimate stress).

   b. Energy-Absorbing Material

   $\epsilon = 0.8\epsilon_u$

where:

ε_u = maximum crushable height at uniform crushable strength.

5. A dynamic increase factor is used for steel which is designed to remain elastic.

3.6.2.4 Guard Pipe Assembly Design Criteria

Protective assemblies/guard pipes are not employed in this design.

3.6.2.5 Material to be Submitted for the Operating License Review

This will be provided in RESAR-SP/90 FDA version.

3.6.2.6 References

1. ASME Section III, Subsection NB and NC-3650.

2. "MULTIFLEX, A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," WCAP-8708 (Westinghouse Proprietary Class 2), February 1976, and WCAP-8709 (Westinghouse nonproprietary), February 1976.

3. "Documentation of Selected Westinghouse Structural Analysis Computer Codes," WCAP-8252, Revision 1, (Westinghouse), May 1977.

4. Moody, F. J., Fluid Reaction and Impingement Loads, Paper Presented at the ASCE Specialty Conference, Chicago, December 1973.

5. "Simplified Pipe Whip Analysis and Restraint Design Procedures," WCAP-10221, December 1982.

TABLE 3.6-1 (SHEET 1 of 2)

ESSENTIAL AND HIGH-ENERGY SYSTEMS

System	Essential[a] Systems	High[b] Energy
Reactor Coolant	X	X
Component cooling water	X	-
Emergency core cooling	X	X
Residual heat removal	X	-
Containment spray	X	-
Chemical volume and control	X	X
Nuclear sampling	-	X
Spent fuel cooling and purification	-	-
Main steam	X	X

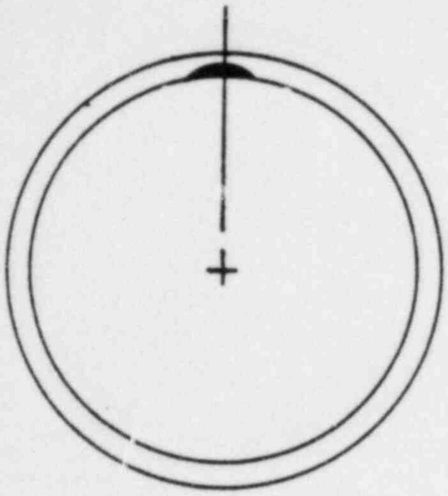
a. Not all essential systems are required for all postulated piping failures. Also, not all portions of essential systems are required for a postulated piping failure.

b. Not all portions of high-energy systems contain high-energy fluid.

TABLE 3.6-1 (SHEET 2 of 2)

ESSENTIAL AND HIGH ENERGY SYSTEMS

System	Essential[a] Systems	High[b] Energy
Main feedwater	X	X
Emergency Feedwater	X	X
Steam generator blowdown	X	X
Safety-related heating, ventilating, and air conditioning	X	-
Essential chilled water	X	-
Waste processing	-	X
Diesel generator and related systems	X	X
Fire protection	-	-
Instrument and service air.	-	-

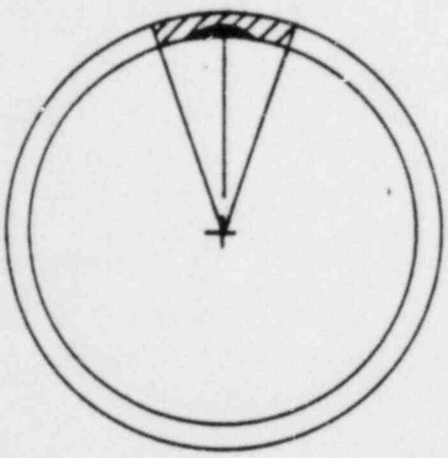


(a) - Postulated Fabrication Flaw

- Select locations in the pipe to be considered based on highest stress for each material location.

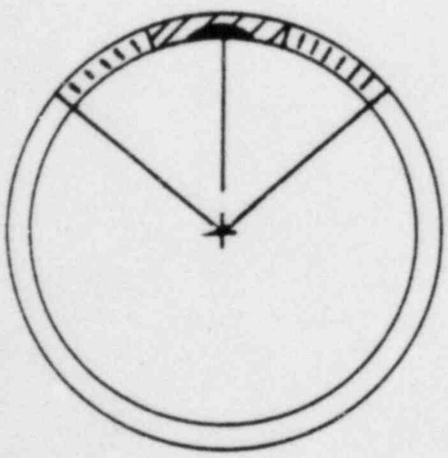
- Postulated crack that may be missed during fabrication and pre-service inspections or would be permitted by Code, whichever is larger.

- Demonstrate by analysis that crack will not grow significantly during service.



(b) - Postulated Leakage Crack

- Assume a crack which allows leakage that is 2 times greater than minimum leak detection capability under normal operating loads so that detection of crack is assured.



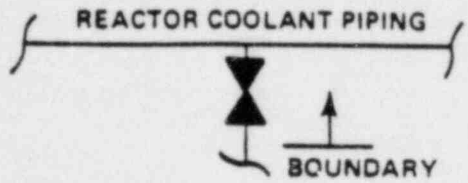
(c) - Stability and Critical Crack Size

- Compare postulated leakage crack size to critical crack size under normal plus earthquake loads.

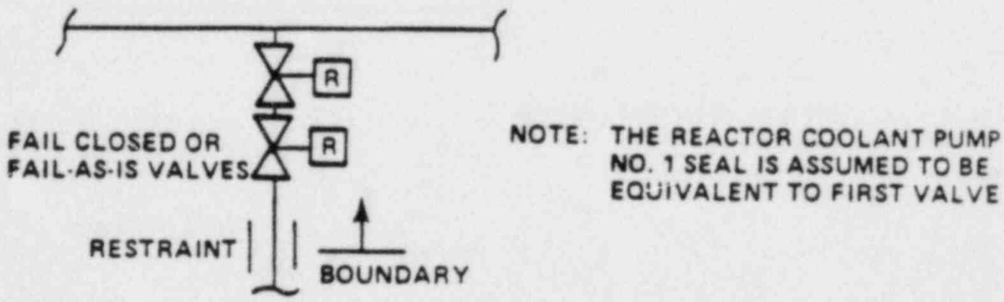
- Demonstrate that 2 times the postulated leakage crack size is stable and, thus, less than the critical crack size.

Figure 3.6-1. Analysis Criteria for Mechanistic Pipe Break Approach

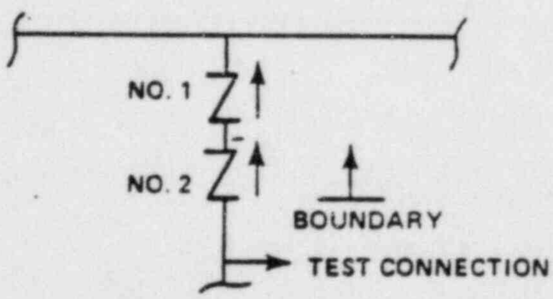
CASE I OUTGOING LINES WITH NORMALLY CLOSED VALVE



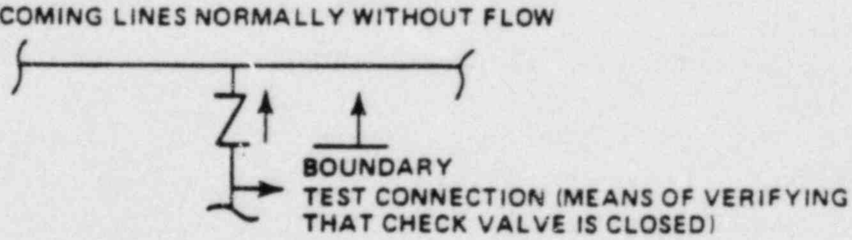
CASE II OUTGOING LINES WITH NORMALLY OPEN VALVES



CASE III INCOMING LINES NORMALLY WITH FLOW



CASE IV INCOMING LINES NORMALLY WITHOUT FLOW



CASE V ALL INSTRUMENTATION TUBING AND INSTRUMENTS CONNECTED DIRECTLY TO THE REACTOR COOLANT SYSTEM IS CONSIDERED AS A BOUNDARY. HOWEVER, A BREAK WITHIN THIS BOUNDARY RESULTS IN A RELATIVELY SMALL FLOW WHICH CAN NORMALLY BE MADE UP WITH THE CHARGING SYSTEM.

Figure 3.6-2. Loss of Reactor Coolant Accident Boundary Limits

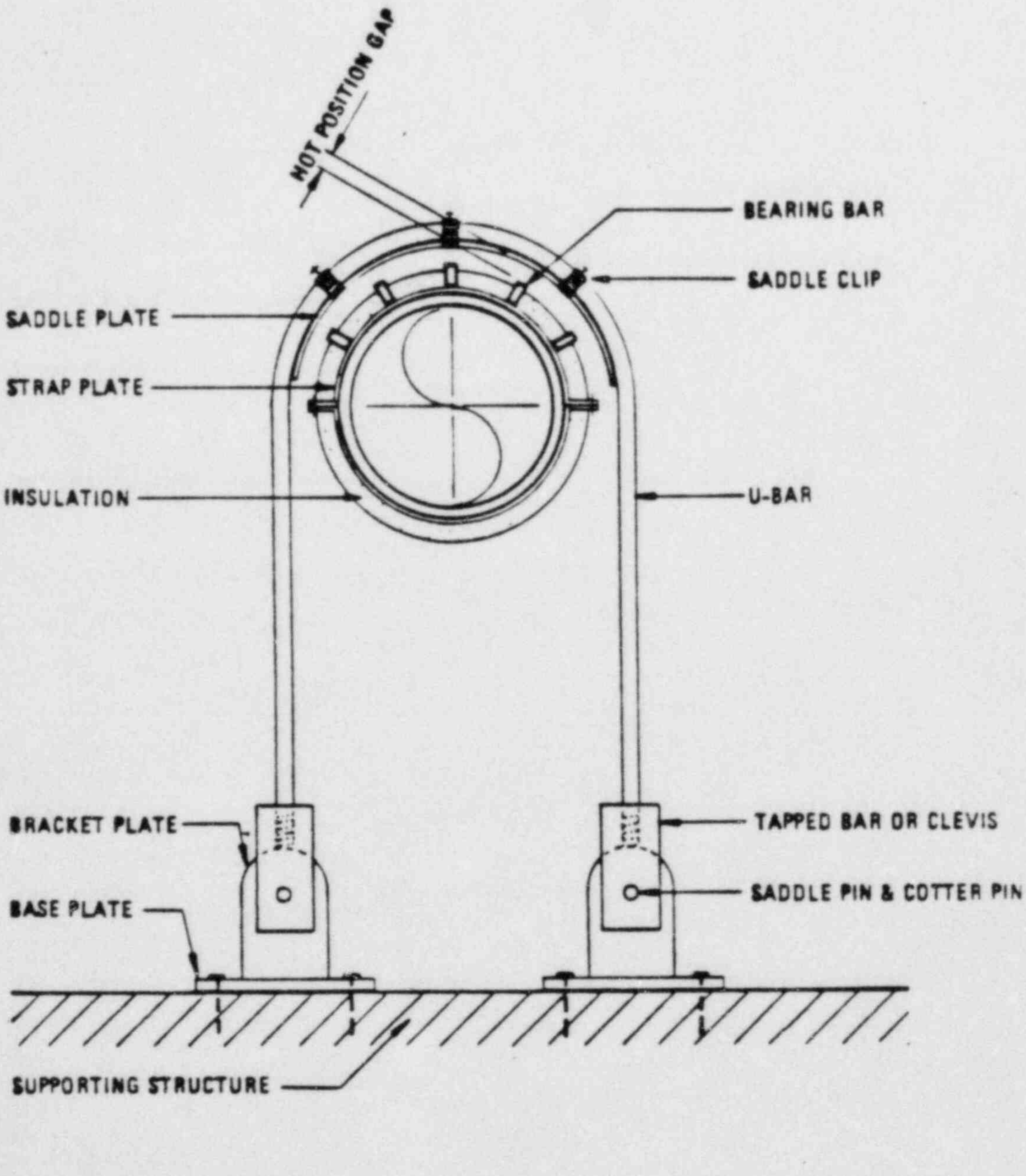


Figure 3.6-3. Typical U-Bar Restraint

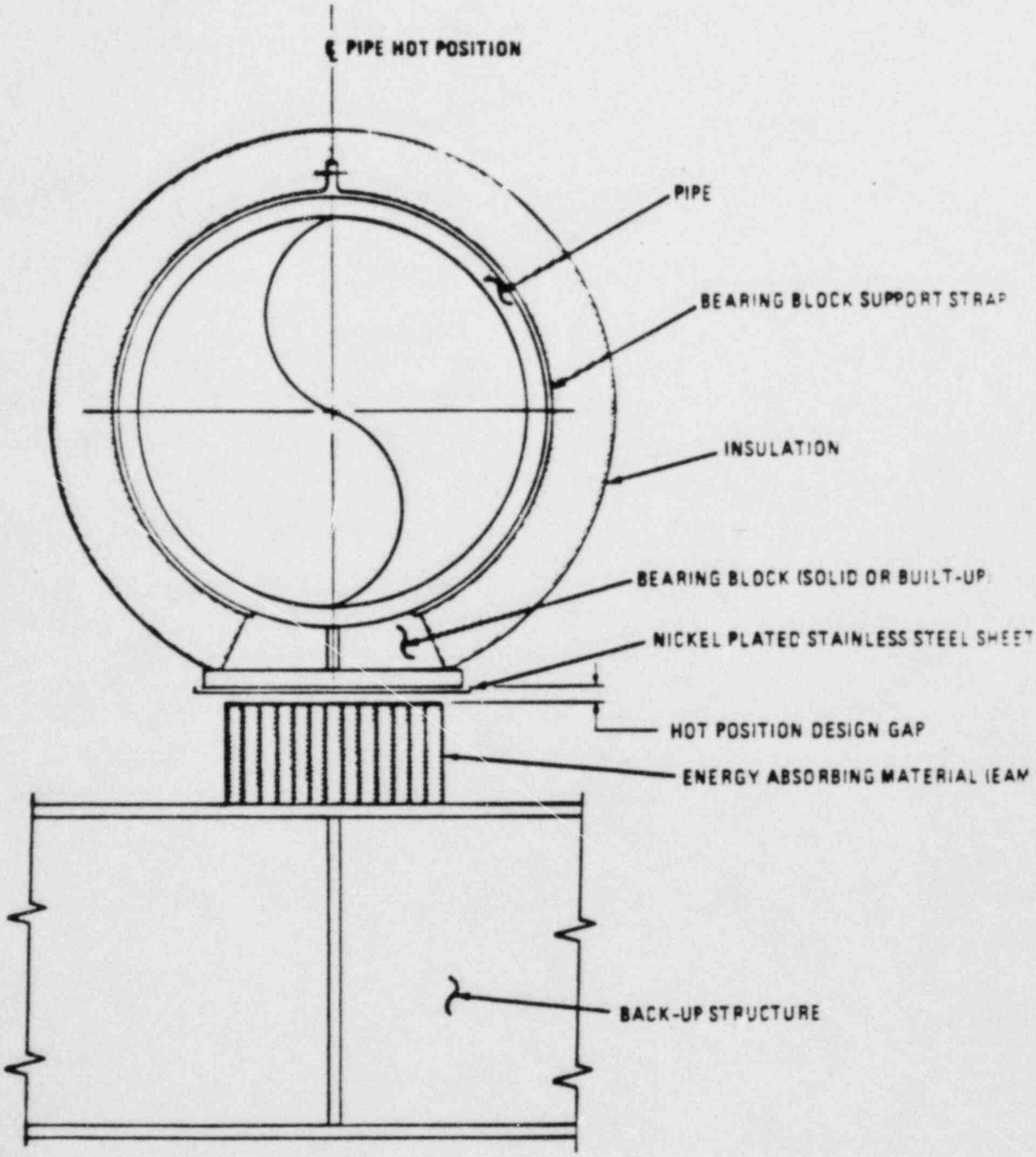


Figure 3.6-4. Typical EAM Restraint

3.7 SEISMIC DESIGN

The seismic design requirements vary in accordance with the seismic classification of structures, systems and components.

The seismic loading requirements are characterized by the safe shutdown earthquake (SSE) and the operating basis earthquake (OBE). The SSE is defined as the maximum vibratory ground motion at the plant site that can be reasonably predicted from geologic and seismic evidence. The OBE is that earthquake which, considering the local geology and seismology, can be reasonably expected to occur during the plant life.

All seismic Category I structures, systems and components are designed for SSE and OBE conditions. The structures, systems and components that are not sufficiently separated by distance or by barriers such that their failure could result in loss of a required safety function are classified as seismic Category II. Seismic Category II structures, systems and components are designed together with their supports to maintain their structural integrity during the SSE.

3.7.1 Seismic Input

As described in Subsection 2A.5.2 of Appendix 2A, RESAR-SP/90 PDA Module 3, "Introduction and Site", a ZPA of 0.1 g for OBE and 0.3 g for SSE free field input motions are established as the baseline seismic condition for the NPB design application. Figure 3.7-1 shows the ground response spectrum normalized at 0.3 g ZPA of an SSE free field input for the NPB. The broad frequency band for spectral amplification and the relatively high ZPA amplitude of 0.3 g are expected to cover a variety of site application conditions. Section 2A-5 of RESAR-SP/90 PDA Module 3, "Introduction and Site" and Subsection 3.7.2.4 describe the use of the seismic input in conjunction with a wide range of foundation properties for soil-structure interaction analysis to characterize the envelop seismic input to structures, systems and components. As a result, when the floor response spectra for a specific plant site are generated through a final design verification analysis, the plant specific floor response spectra at key locations will be properly enveloped by the floor response spectra developed for the NPB design.

3.7.1.1 Design Response Spectra

The free-field design response spectra are shown in Figures 3.7-1 and 3.7-2 for the horizontal and the vertical components of the SSE and in Figures 3.7-3 and 3.7-4 for the horizontal and vertical components of the OBE. The design response spectra are in conformance with Regulatory Guide 1.60, Design Response Spectra for Seismic Design of Nuclear power plants.

3.7.1.2 Design Time-History

Synthesized time histories of 10-second total duration are generated for each of the three components - two horizontal and one vertical, of the SSE seismic design response spectra of Subsection 3.7.1.1. The time histories are normalized in each direction at 0.3 g for SSE and 0.1 g for OBE.

As shown in Figures 3.7-5 through 3.7-7, the design response spectra are properly enveloped by the response spectra calculated for the synthesized time histories in all three directions. The selection of frequency intervals for response spectra calculation is consistent with that of Regulatory Guide 1.122.

3.7.1.3 Critical Damping Values

Damping is an energy dissipation mechanism which reduces the amplification of the vibratory response. Critical damping is defined as the least amount of damping which causes a single-degree-of-freedom viscous system to return to its original position without oscillation after initial disturbance. For analytical modeling purposes, damping ratio shown as a percentage or fraction of critical damping value is often specified to account for a variety of energy dissipation mechanisms which can be related to material, response condition and type of connections of a structure. For typical structures and components, the damping values are considered as given in Table 3.7-1.

Consistent with the Westinghouse positions in Reference 1, the NPB design uses the damping values of 8% and 5% of critical for the respective SSE and OBE events for the primary coolant loop systems. For the remaining safety related piping systems, the frequency dependent damping values as established by the Pressure Vessel Research Council Technical Committee on Piping Systems and endorsed by Westinghouse in Reference 1 are considered. Figure 3.7-8 summarizes the damping values used for the piping design/analysis for the NPB.

Also consistent with Regulatory Guide 1.61, damping values higher than those cited above may be used if justified by test results. Using the tests reported in References 2, 3 and 4, the OBE and SSE damping ratios of 7 percent and 10 percent, respectively, will be considered for the fuel assembly and a damping ratio of 5 percent will be specified conservatively for both the OBE and SSE seismic response analyses for the control rod drive mechanism (CRDM).

3.7.1.4 Supporting Media for Seismic Category I Structures

Refer to Subsection 2A.5.3 of RESAR-SP/90 PDA Module 3, "Introduction and Site".

3.7.2 Seismic System Analysis

This subsection describes the seismic analyses of the Category I structures, systems and components. Seismic systems are defined herein as the Category I structures, systems or component which, for analysis purposes, are considered in conjunction with foundation medium in forming a soil-structure or foundation-structure interaction model. All Category I structures, systems and components not designed as seismic systems and all Category I distributive systems such as heating, ventilation and air-conditioning systems, electrical cable trays, conduits, and piping are considered as seismic subsystems and their analyses are described in Subsection 3.7.3 of this module.

3.7.2.1 Seismic Analysis Methods

Seismic systems are analyzed by direct integration to determine the effects of input ground motions on the Nuclear Power Block to obtain structural design loads for the seismic systems and to define seismic environment for the subsequent seismic analysis of structures, systems and components which are not supported directly on soil or foundation medium.

The analyses of the seismic systems are performed on the baseline configuration of the Nuclear Power Block to resist the ground motions of 0.1 g and 0.3 g ZPA for the respective OBE and SSE as defined in Subsection 3.7.1. The potential variability of any site specific soil condition as defined in Section 2A-5 of RESAR-SP/90 PDA of Module 3, "Introduction and Site" is covered by this baseline design as a result of employing the envelope seismic requirements which are resulting from using bounding soil properties in the soil-structure interaction analyses. As discussed in Subsection 3.7.2.4, the envelope seismic requirements are derived by considering three analysis cases characterized by the soil shear wave velocity of 1000 ft/sec, 2500 ft/sec and infinite each in the half-space impedance function modeling method of the soil-structure interaction models. By performing seismic analyses as described in the following paragraphs, the seismic performance of the baseline configuration is expected to be more than sufficient when the site specific data are incorporated into the final design verification by the impedance function method and the finite boundaries modeling methods to demonstrate the design adequacy as required by the Standard Review Plan (NUREG-0800, Rev. 1, July 1981).

3.7.2.2 Natural Frequencies and Response Loads

The seismic system analyses of the building structure of the NPB are performed using a time-history direct integration method. The floor response spectra generated from this method is indicative of the frequency content of the soil-structure system. Natural frequencies, mode shapes or modal responses are not obtained in this method as in the response spectrum analysis method.

The direct integration method provides the internal force time history responses for the sticks used to model the NPB building. Seismic design load information such as uplift force, shear, torsion and overturning moment are computed. Alternatively, seismic design loads may be obtained using the response spectrum analysis methods described in Subsection 3.7.3.

3.7.2.3 Procedure Used for Modeling

The Category I seismic systems are modeled for seismic analysis by appropriately accounting for the effects of soil-structure interaction to simulate the overall behavior of the seismic systems. Figure 3.7-9 displays the stick model representation of the seismic systems to model the mass and stiffness properties of the reactor external building, the concrete shield building, interior concrete shield, and the steel spherical containment fastened on the common base mat. The discrete masses are lumped at the nodes located at the floor levels and the locations of major discontinuity of the building systems. Six degrees of freedom are assigned to each node although only three translational masses and one rotatory mass moment of inertia about the vertical axis are considered for each node. The equivalent three-dimensional beam elements are used to connect the nodal points to model the stiffness relationship between nodes.

To account for the soil-structure interaction effect, the building stick model is coupled to the discrete soil dynamic properties through a common nodal point of the mass center of the base mat. At this common node, three translational and three rotational degrees of freedom as depicted in Figure 3.7-9 are introduced. As discussed in Subsections 3.7.2.4 and 3.7.2.5, this soil-structure interaction model is used to generate the seismic loads on structure and the floor response spectra of the buildings.

3.7.2.4 Soil-Structure Interaction

The acceptance criteria of Subsection 3.7.2 of the SRP require that modeling methods for conducting soil-structure interaction analysis include both the half-space impedance method and the finite boundaries methods. This requirement for design verification analysis will be accomplished once the site soil data are provided by the applicants and a final analysis is done to confirm the design.

The soil-structure interaction effect is dependent on the site specific configuration such as embedment depth, dynamic nonlinear soil behavior, depth and boundaries of soil layering and the analysis techniques employed. The purpose of combining all these parameters in a soil-structure interaction analysis is to simulate the dynamic properties of the soil-structure interaction system. In recognizing the potential effects of these parameters, the impedance function model is considered sufficient for the purposes of generating the seismic structural design load information and the bounding seismic input to the subsystems of the baseline configuration. The following modeling considerations as outlined in the solid blocks of Figure 3.7-10 for the soil-structure interaction analysis are sufficient for the Nuclear Power Block generic applications.

1. The stick models represent the baseline configuration of the NPB seismic systems including the reactor external building, the shield building, the interior concrete and the steel containment.

2. The equivalent stiffness and damping coefficients of a soil medium are characterized by three types of soil at a shear wave velocity of 1000 ft/sec, 2500 ft/sec and infinite to envelope the potential site specific variability of soils. The procedures to evaluate the stiffness and damping properties of the soil media are in accordance with those of the "Standard for the Seismic Analysis of Safety-Related Nuclear Structures", ASCE (Reference 5).

3. The building and the soil analytical models are coupled to determine the responses of the systems. These responses include uplift force, shears, torsion, displacements and overturning moments at the individual floors and the interface between the building and the base mat.

4. The soil-structure interaction model is used in a time-history analysis to compute the in-structure response spectra for all the important locations where the subsystems will be located and seismic analyses are required to demonstrate the seismic performance of subsystems. As described in Subsection 3.7.2.5, response spectra at three critical locations are selected to define the interface site specific requirements to demonstrate design adequacy of the NPB application.

As indicated in Figure 3.7-10, the baseline configuration of the NPB is qualified for the generic design ground motion. During the design verification for the site specific application, the site specific requirements will be evaluated against the envelope seismic capabilities estabished during the qualification of the baseline NPB configuration. Because of the inherent margin for the generic design of the NPB, the final design verification for soil-structure interaction analysis can be accomplished without the need for detailed seismic analysis of structures, systems and components which have been qualified for the baseline configuration.

3.7.2.5 Development of Floor Response Spectra

The floor response spectra for the NPB buildings are developed using the generic time-history analysis for the soil-structure models. Time-history responses are obtained at floors for three orthogonal directions considering concurrent horizontal and vertical seismic inputs to the analysis models. The frequency intervals used for computing the spectra are consistent with those of Regulatory Guide 1.122. Since the natural frequencies are not computed for the soil-structure system and since the intervals between the selected frequencies are small, no additional frequency points are specifically identified for calculating response spectral values.

Figures 3.7-11 through 3.7-19 display the results of floor response spectra calculated for the three critical interface locations: 1) the reactor pressure vessel support, 2) the operating deck of the reactor containment structure, and 3) the control room floor of the reactor external building. In each figure which represents spectral requirements in one designated direction, the 5% damping generic floor response spectrum drawn with a solid line is constructed to envelop the three floor response spectra resulting from the analytical models of three different soil conditions. In accordance with Regulatory Guide 1.122, the spectrum band-broadening of 15% is introduced. The generic floor response spectra serve as a basis for designing the subsystems of the baseline configuration of the NPB. The site specific floor response spectrum obtained will be measured against these generic floor response spectra during the final verification analysis of the NPB.

3.7.2.6 Three Components of Earthquake Motion

The soil-structure systems are analyzed with the three-dimensional models subjected to simultaneous input seismic motions in three orthogonal directions. As a result, the three-component earthquake effects need not be addressed.

3.7.2.7 Combination of Modal Responses

This subsection does not apply as only direct integration time-history analyses are performed on the systems.

3.7.2.8 Interaction of Non-Category I Structures With Seismic Category I Structures

The non-category I structures are designed to resist seismic effects as discussed in Section 3.7. The deflections of the Category I structures are provided for the design of the non-category I structures for preventing potential impacting between adjacent buildings. The soil-structure interaction between the turbine generator building and its underlying soil may affect the seismic behavior of the adjacent NPB structure system. However, the effects of this shallowly embedded structure are judged to be small and are expected to be covered by the present use of the enveloping soil properties data for the response calculation for the NPB.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

The uncertainties associated with the analytical models for deriving the in-structure response spectra are accounted for by broadening the spectrum peaks by $\pm$15% across the frequency band in accordance with Regulatory Guide 1.122. As described in Subsection 3.7.2.5, each generic spectrum curve envelopes the broadened spectrum peaks for the three soil-structure analytical models. The resulting broad-band envelope floor response spectra are more than sufficient to address the potential uncertainties in modeling the seismic systems.

3.7.2.10 Use of Constant Vertical Static Factors

No constant vertical static factors are used for Category I structures. The same method of analysis is used for both vertical and horizontal responses of the structures.

3.7.2.11 Method Used to Account for Torsional Effects

The development of the generic floor response spectra by varying the underlying soil properties contains sufficient margin to account for the accidental torsional effects on a site specific configuration. The generic floor response spectra are judged to be sufficient for analysis and design of subsystems anchored on floors.

3.7.2.12 Comparison of Responses

This section does not apply as response spectrum analysis will not be utilized for the systems of the NPB. Response spectra comparison, however, will be used to demonstrate the conservatism of the generic baseline design as applied to site specific configuration.

3.7.2.13 Methods for Seismic Analysis of Dams

Not applicable to the NPB scope.

3.7.2.14 Determination of Seismic Category I Structure Overturning Moment

The Category I NPB structure is designed to resist overturning due to the combined effects of the vertical and two horizontal components of seismic ground motion. The moment equilibrium method is considered in which the maximum seismic overturning moment is obtained from the analyses described in Subsection 3.7.2.2. The gravity force reduced by the hydrostatic buoyance force provides stability of the structure in resisting overturning moment. The minimum safety factor against overturning moment is 1.1 for an SSE combined with the other applicable loading conditions.

3.7.2.15 Analysis Procedure for Damping

The damping property of the soil-structure system of the NPB is affected by the type of soil medium and the details of the structural concrete and steel used for constructing the plant. In order to model the system dynamic properties the ASCE nonproportional damping modeling approach of Reference 5 is considered. This is accomplished by introducing first the segmentally proportional damping for the respective concrete and steel portions of the NPB building structures for the stick model of Figure 3.7-9. For the soil medium, a set of discrete dampers and springs are introduced at the base of the building structures in accordance with the impedance function approach of Reference 5. This resulting nonproportionally damped soil-structure system calls for dynamic response analysis by the direct integration approach as discussed in Subsection 3.7.2.1.

3.7.3 Seismic Subsystem Analysis

This section describes the seismic analysis performed on subsystems.

3.7.3.1 Seismic Analysis Methods

Both the time-history solution and the response spectrum analysis technique are used for analyzing the subsystems of the NPB. In general, analyses follow the ASCE Seismic Analysis Standard Committee approaches of Reference 5, "Standard for the Seismic Analysis of Safety-Related Nuclear Structures."

The generic floor response spectra of Subsection 3.7.2.5 serve as design input for the subsystems.

When the time-history solution is considered, synthesized time histories of 10-second total duration are generated for each of the three components - two horizontal and one vertical, of the floor response spectra. The Westinghouse program DEBLIN2, Reference 6, is utilized to synthesize the spectrum-compatible time histories. The program modifies earthquake motions by a frequency suppressing and raising technique in an iterative scheme to assure that the response spectra of the resulting time histories will properly envelope the corresponding floor response spectra. Statistical independence among the time history components is assured by requiring the cross correlation coefficients among different inputs to be less than 0.3 (References 7 and 8). The resulting three components of the acceleration time histories will be simultaneously input to subsystems for either a direct-integration or a modal superposition time-history solution.

For subsystems modeled with linear elastic response, the response spectrum analysis of Reference 5 is performed. The generic floor response spectra are applied to subsystems with consideration of the three components of earthquake motion as per Subsection 3.7.3.6 and the combination of modal responses as per Subsection 3.7.3.7.

3.7.3.2 Determination of Number of Earthquake Cycles

For each OBE the system and component will have a maximum response corresponding to the maximum induced stresses. The effect of these maximum stresses for the total number of OBE's must be evaluated to assure resistance to cyclic loading.

The OBE is conservatively assumed to occur five times over the life of the plant. The number of maximum stress cycles for each occurrence depends on the system and component damping values, complexity of the system and component, duration and frequency contents of the input earthquake. A precise determination of the maximum number of stress cycles can only be made using time history analysis for each item which is not feasible. Instead, a time history study has been conducted to arrive at a realistic number of maximum stress cycles for all Westinghouse systems and components.

To determine the conservative equivalent number of cycles of maximum stress associated with each occurrence, an evaluation was performed considering both equipment and its supporting building structure as single degree-of-freedom systems. The natural frequencies of the building and the equipment are conservatively chosen to coincide.

The results of this study indicate that the total number of maximum stress cycles in the equipment having peak acceleration above 90 percent of the maximum absolute acceleration did not exceed eight cycles. If the equipment was assumed to be rigid in a flexible building, the number of cycles exceeding 90% of the maximum stress was not greater than three cycles.

This study was conservative since it was performed with single degree-of-freedom models which tends to produce a more uniform and unattenuated response that a complex interacted system. The conclusions indicate that 10 maximum stress cycles for flexible equipment (natural frequencies less than 33 Hz.) and 5 maximum stress cycles for rigid equipment (natural frequencies greater than 33 Hz) for each of 5 OBE occurrences should be used for fatigue evaluation of WAPWR systems and components.

3.7.3.3 Procedure Used for the Modeling

A. Modeling of Piping Systems for Dynamic Analysis

The piping systems are modeled utilizing a three-dimensional structural representation composed of concentrated lumped masses connected by appropriate piping system. The model accounts for the interaction effect between piping, equipment and supports. Supports are modeled as flexible members with the appropriate stiffness to represent the support compliance. The piping model is terminated at equipment nozzles which are modeled as rigid anchors with consideration given to the seismic amplification of equipment, as follows:

1. For rigid equipment in which the fundamental frequency is equal or higher than 33 Hz, the amplified response spectra of the structure is used.

2. For equipment in which the fundamental frequency is lower than 33 Hz, the amplified response spectra and the seismic anchor displacement of the equipment at the pipe/nozzle interface point is used. Alternatively, a simplified model of the equipment to account for dynamic interaction and amplification is coupled with the piping model, and the amplified structure response spectra are used to excite the coupled model.

All in-line components are included in the model. The concentrated mass of in-line components such as valves, flanges, and strainers are represented as lumped masses. Valve operators are modeled as an offset lumped mass to account for the torsional and in-plane bending effects on the piping.

The following criteria are used for the decoupling of piping subsystems:

1. When piping is decoupled from the equipment, the nozzle is modeled as a full, six-degree-of-freedom restraint.

2. For the analysis of main runs, branch connections are decoupled from the main runs when the ratio of the branch to run section moduli is equal to or less than 1/16, or the ratio of the branch to run moment of inertia is 1/50.

3. Piping subsystems (main or branch runs) which are decoupled into separate analytical models satisfy one of the following criteria:

(a) The decoupling point is a full anchor for the piping of both separate models.

(b) The boundary of each decoupled model contains a sufficiently long region of common overlap to other models which effectively provides restraint(s) in each of the three orthogonal directions in order to justify decoupling.

B. Modeling of Equipment

Seismic Category I equipment is modeled as lumped systems which consist of a series of discrete mass points connected by elastic members. All significant concentrated weights are represented as lumped masses. Typical examples of concentrated weights are weights of motor rotor and pump impeller in the analysis of shafts. The number of dynamic degrees of freedom is at least twice the number of modes having frequencies less than 33 Hz.

3.7.3.4 Basis for Selection of Frequencies

There are no specific design criteria that attempt to cause the fundamental frequencies of NPB equipment to be different from the forcing frequencies of the supporting structures. The effect of the equipment fundamental frequencies relative to the supporting structure forcing frequencies is, however, considered in the analysis of the NPB equipment.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

The static load equivalent or static analysis method involves the multiplication of the total weight of the equipment or component member by the specified seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient is established on the basis of the expected dynamic response characteristics of the component. Components that can be adequately characterized as single-degree-of-freedom systems are considered to have a modal participation factor of 1. Seismic acceleration coefficients for multidegree-of-freedom systems may be determined as 1.5 times the peak spectral acceleration of the applicable response spectrum. Smaller values may be used, if justified.

3.7.3.6 Three Components of Earthquake Motion

Seismic Category I subsystems and components are analyzed by considering the combined effects of seismic loads occurring in three mutually perpendicular directions, two in the horizontal direction and one in the vertical direction. The total combined response (displacements, stresses, and forces) due to the three components of earthquake motion is obtained by using the square-root-sum-of-the-squares (SRSS) formula applied to the resultant codirectional responses. For instance, for each item of interest, such as displacement, force, stress, etc., the total response is obtained by applying the SRSS method. The mathematical expression for this method (with R as the item of interest) is:

$$R_C = \left(\sum_{T=1}^{3} R_T^2\right)^{1/2} \qquad (1)$$

where:

R_C = total combined response at a point.
R_T = value of combined response of direction T.

The system and equipment response can also be determined using time-history analyses. When a time-history analysis is performed, the two horizontal and the vertical time-history components are applied simultaneously.

3.7.3.7 Combination of Modal Responses

The total codirectional seismic response is obtained by combining the individual modal responses utilizing the SRSS method. An optional method is the algebraic combination of modes with closely spaced frequencies (Reference 9).

The groups of closely spaced modes are chosen such that the difference between the frequency of the first mode and the last mode in the group does not exceed 10 percent of the lower frequency. Groups are formed starting from the lowest frequency and working toward successively higher frequencies. No one frequency is in more than one group. The combined total response is obtained as follows:

$$R_T2 = \sum_{i=1}^{N} R_i^2 + 2 \sum_{j=1}^{S} \sum_{K=M_j}^{N_j-1} \sum_{\ell=K+1}^{N_j} R_K R_\ell \varepsilon_{K\ell} \tag{2}$$

where:

R_T = total codirectional response.

R_i = response of mode i.

N = total number of lower frequency, flexible modes.

S = number of groups of closely spaced modes.

M_j = lowest modal number associated with group j of closely spaced modes.

N_j = highest modal number associated with group j of closely spaced modes.

$\varepsilon_{K\ell}$ = coupling factor with

$$\varepsilon_{K\ell} = [1 + (\frac{\omega'_K - \omega'_\ell}{(\beta'_K\omega_K + \beta'_\ell\omega_\ell)})^2]^{-1} \quad \text{and} \tag{3}$$

$$\omega'_K = \omega_K \ (1 - (\beta_K)^2)^{1/2} \tag{4}$$

$$\beta'_K = \beta_K + \frac{2}{\omega_K \ t_d} \tag{5}$$

where:

ω_K = frequency of mode K.
β_K = fraction of critical damping in mode K.
t_d = duration of the earthquake.

The options used to account for high-frequency (>33.0 HZ) modes are described below:

A. The Residual Load Method (RLM) with Uniform Response Spectrum Analysis is based on the following equations (Reference 10):

$$\{X_c\} = -[K]^{-1} [M]([J]-[\phi_d] [\phi_d]^T [M] [J]) \{\ddot{X}_g\} \quad (6)$$

$$\{\ddot{X}_c\} = ([J] - [\phi_d] [\phi_d]^T [M] [J]) \{\ddot{X}_g\} \quad (7)$$

where

[J] = influence matrix
[K] = stiffness matrix
[M] = mass matrix
$\{X_c\}$ = residual displacement vector from truncated higher modes
$\{\ddot{X}_c\}$ = residual acceleration vector
$\{\ddot{X}_g\}$ = ground acceleration vector
$[\phi_d]$ = flexible mode shape matrix

The combination of shock directions for these truncated higher modes is obtained from equation (1). The total response from flexible and truncated higher modes is given by:

$$R_{CTOTAL} = (R_{CF}^2 + R_{CT}^2)^{1/2} \quad (8)$$

where R_{CF}, R_{CT} are the combined flexible and truncated mode responses from equation (1).

B. The full zero period acceleration method (FZPA) with Uniform Response Spectrum Analysis is based on the following equations (Reference 10):

$$\{X_c\} = -[K]^{-1} [M] [J] \{\ddot{X}_g\} \tag{9}$$

$$\{\ddot{X}_c\} = [J] \{\ddot{X}_g\} \tag{10}$$

The combination of shock directions for the FZPA response is obtained from equation (1). The total response from flexible modes and the FZPA response is obtained by SRSS combination similar to equation (8).

C. The RLM with Multiple Response Spectrum Analysis is based on the following equations (Reference 11):

$$\{X_c\}= -[K]^{-1} [M]([Y]-[\Phi_d] [\Phi_d]^T [M][Y]) \{\ddot{X}_g\} \tag{11}$$

$$\{\ddot{X}_c\}= ([Y] - [\Phi_d] [\Phi_d]^T [M] [Y]) \{\ddot{X}_g\} \tag{12}$$

where

$$[Y] = -[K]^{-1} [K_G] \tag{13}$$

and K_G is system-support coupling stiffness matrix,

The combination of shock directions along with flexible and truncated modes is performed in the same manner as in item A above. (See subsection 3.7.3.9 for further details).

D. The FZPA method with Multiple Response Spectra Analysis is based on the following equations (Reference 11):

$$\{X_c\} = -[K]^{-1} [M] [Y] \{\ddot{X}_g\} \tag{14}$$

$$\{\ddot{X}_c\} = [Y] \{\ddot{X}_g\} \tag{15}$$

The combination of shock directions along with flexible and truncated modes is performed in the same manner as in item B above. (See Subsection 3.7.3.9 for further details.)

3.7.3.8 Analytical Procedures for Piping

The Seismic Category I piping systems are analyzed and evaluated according to the rules of the American Society of Mechanical Engineers (ASME) code, Section III. When modal seismic response spectrum analysis methods are used to evaluate piping seismic response due to inertial loading arising from excitations at different supports within one or more buildings, the procedures described in Section 3.7.3.9 are used. The effect of differential seismic anchor motions at different supports are included in the piping analysis according to the rules of the ASME code, Section III. The piping stresses due to seismic anchor motions are combined with stresses from other applicable loads including seismic inertial loading and then evaluated as required by the ASME code, Section III. For analysis of seismic anchor motions, the procedures described in Section 3.7.3.9 are used.

3.7.3.9 Multiple Supported Equipment Components With Distinct Inputs

A. To evaluate piping and equipment components seismic response due to inertial loading arising from excitations at different supports within one or more buildings, either modal envelope seismic response spectra analysis method or modal non-uniform seismic response spectra analysis method is used.

A.1 The modal envelope seismic response spectra analysis method is the same as the standard model seismic response spectra analysis method for a singly- or multiply-supported system subject to uniform translational seismic excitation except that it utilizes, for each direction of excitation the single envelope spectrum or the worst single spectrum. The single envelope or worst spectrum is assumed by this procedure to account for the influence of phasing and interdependence characteristics of non-uniform excitation represented by translational spectra at various supports.

A.2 In the modal non-uniform seismic response spectra analysis method (References 11, 12 and 13), for each direction of excitation, multiple input response spectra representing the non-uniform seismic excitation at all support (boundary) points of the structural system (model) are explicitly used without being approximated, consolidated or replaced as in the case of the modal superposition envelope response spectra method. Further, for each direction of excitation, the phasing and interdependence characteristics of the multiple input response spectra representing non-uniform seismic excitation are identified and properly accounted for by this method as outlined below.

Proportional Input - For this type of input, the support motion at a given point can be obtained simply through multiplication of a reference excitation by a real number. This, therefore, includes the uniform excitation as a special case. Support motions that are 180° out-of-phase are also included here since they can be obtained through multiplication of a reference excitation by a negative real number. Support point motions associated with a single-mode response of a supporting structure or with a rigid supporting structure are examples of this type of input. For this type of input, the representative maximum modal response is obtained by algebraic combination of contributions of individual support point inputs.

Independent Input - For this type of input, the support motions are treated to be statistically independent and are therefore essentially uncorrelated. Support point motions associated with supporting structures of widely differing dynamic characteristics can be considered as practical examples of this type of input. For this type of input, the representative maximum modal response is obtained by the square-root-sum-of-squares (SRSS) combination of contributions of individual support point inputs.

Mixed Input - This type of non-uniform excitation consists of a combination of the two types described above. For this type of input, the representative maximum modal response is obtained by the SRSS combination of contributions of the representative maximum modal responses obtained for each of the two types described above.

After maximum possible use of algebraic and SRSS combinations, as described above, absolute sum combination is used only as a last resort in absence of another more realistic combination.

B. The response due to differential seismic anchor motions is calculated using static analysis (without including dynamic load factor). In this analysis, the static model is identical to the static portion of the dynamic model used to compute the seismic response due to inertial loading. In particular, the structural system supports in the static model are identical to those in the dynamic model. The effect of relative seismic anchor displacements are obtained either by using the worst combination of the peak displacements or by proper representation of the relative phasing characteristics associated with different support inputs.

C. The results of modal seismic spectra analysis in Item A above and the results of seismic anchor motion analysis in Item B above are combined by the SRSS when required by consideration for the ASME classification of the stresses.

3.7.3.10 Use of Constant Vertical Static Factors

Constant vertical load factors are not used as the vertical floor response load for the seismic design of safety-related components and equipment within the Westinghouse scope of responsibility.

3.7.3.11 Torsional Effects of Eccentric Masses

The effect of eccentric masses, such as valves and valve operators, is considered in the seismic piping analyses. These eccentric masses are modeled in the system analysis, and the torsional effects caused by them are evaluated and included in the total system response. The total response must meet the limits of the criteria applicable to the safety class of the piping.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels

There are no buried seismic Category I piping systems and tunnels in the NPB.

3.7.3.13 Interaction of Other Piping With Seismic Category I Piping

Where seismic Category I piping systems are in close proximity to non-seismic piping, the non-seismic pipes are restrained so that no failure of the seismic Category I system can occur.

Where seismic Category I piping is directly connected to non-seismic Category I piping, the seismic effects of the latter are prevented from being transferred to the seismic Category I piping by use of anchors or a combination of restraints; or when this is not practical, the interactive effects of the unrestrained portion of the non-seismic Category I piping are included in the analyses, and evaluated for acceptability.

3.7.3.14 Seismic Analyses for Reactor Internals (Core, Core Supports, Mechanisms)

Fuel assembly, core support structure, and control mechanism component stresses induced by seismic disturbances, are analyzed by finite element computer techniques. The time-history response of the building is used to generate the input to the system model of the above components. These components are modeled as spring and lumped mass systems or beam elements.

The component seismic response of the fuel assemblies is analyzed to determine design adequacy. The response of the core structures and mechanisms is used in the ASME B&PV code evaluations. Fuel assembly damping and grid strength capability are determined experimentally.

The mechanisms, both the control rod drive mechanism (CRDMs) and the displacer rod drive mechanisms (DRDMs), are seismically analyzed to confirm that stresses under the combined loading conditions, do not exceed allowable levels as defined by the ASME Code, subsection III, for condition B and condition D events. The mechanisms are modeled as a system of lumped and distributed masses, and the resultant seismic bending moments and shear loadings along the length of the mechanisms are calculated. The corresponding stresses are then combined with the stresses from other loadings and the combination is shown to meet the requirements of the ASME Code, Section III.

3.7.3.15 Analysis Procedure for Damping

Where the equipment or component consists of subcomponents with the same damping characteristics, the same critical damping value is used for the entire equipment or component. The corresponding critical damping value is chosen from Table 3.7-1 and Figure 3.7-8. For seismic Category I equipment or component consisting of subcomponents with different damping characteristics, two approaches are considered: 1) the lowest critical damping value associated with the subcomponents in the equipment or component is used in the analysis for all modes, 2) the composite damping values or nonproportional damping models as proposed by the ASCE Seismic Analysis Standard Committee of Reference 5 are used.

3.7.4 Seismic Instrumentation

Seismic instrumentation is provided to the NPB to gather information on the input ground motion and the output vibratory responses of the representative Category I structures and equipment so that an evaluation can be made as to:

o Whether input design response spectra were exceeded,

o Whether the vibratory responses of the representative Category I structures and equipment were exceeded,

o The need for shutdown of the plant, and

- The degree of conservatism of the mathematical models used in the seismic analysis of the building and equipment.

The design consideration of the seismic instrumentation is based on a Safe Shutdown Earthquake (SSE) of 0.3g ZPA and an Operating Basis Earthquake (OBE) of 0.1g ZPA.

3.7.4.1 Comparison With Regulatory Guide 1.12

The seismic instrumentation described below consists of time-history accelerographs, seismic switches, response spectrum recorders and peak accelerographs meeting the USNRC Regulatory Guide 1.12, Revision 1 (April 1974), as required for a severe earthquake with ZPA of 0.3g or higher.

3.7.4.2 Location and Description of Instrument

The instrumentation described below is employed to measure and record the seismic inputs and the plant structural and equipment responses and to provide displays and alarms to operators to act and engineers to evaluate the plant seismic capability after an earthquake.

When external power supply is needed for operating the instrument during and after earthquakes, the Class 1E 120 V uninterruptable power supply will be provided.

3.7.4.2.1 Time-History Accelerograph

Three triaxial time-history accelerographs will be provided, one each at the following locations:

1. A free field at approximately 500 ft from the edge of the reactor external building

2. The top of the foundation base mat (Reference Elevation = 72m) and inside the reactor external building

3. The concrete operating floor (Reference Elevation = 100m) and inside the reactor containment.

A fourth time-history accelerograph will be installed in the main control room concrete floor if the design site ground motion is 0.3g ZPA or higher.

Each time-history accelerograph package consists of a triaxial sensor with triaxial starter unit and a recorder unit. The triaxial sensor unit is responsive in the frequency range of 0.1 Hz to 30 Hz in the three orthogonal axes. The starter unit also has corresponding acceleration sensors set to energize the triaxial sensor unit whenever the threshold acceleration is exceeded in any of the three orthogonal axes. The threshold accelerations are set between 0.005g and 0.02g, depending on locations, to avoid actuation due to insignificant motion, but to record a seismic disturbance which may have a ground acceleration magnitude significantly lower than that of the Operating Basis Earthquake of 0.1 g ZPA.

The recorder units and a common playback unit will be housed in a control panel which in turn will be located in the main control room (Subsection 3.7.4.2.6). The three starter units installed in the main control room, the operating floor and the basemat of the reactor containment will be oriented such that their axes and the axes of the sensor units are pointing in the same direction and aligned to the principal axes of the reactor external building.

The time-history accelerograph is fully operational within 0.1 second of seismic starter actuation. Once actuated, an amber light, one for each accelerograph package, remains on in the control room. The accelerograph will operate continuously during that period in which the acceleration exceeds the starter threshold plus at least five (5) seconds.

The recorder unit will be capable of a minimum 25 minutes total recording time. The common playback system allows immediate graphical time-history accelerogram playback capability.

The starter unit and seismic switch as described in the next paragraph can be tested from the main control room.

3.7.4.2.2 Seismic Switch

One seismic switch will be located at the top of the foundation mat inside the reactor external building and in the general vicinity where the time-history accelerograph of Subsection 3.7.4.2.1 is installed. If the design site specific ground motion is 0.3g ZPA or higher, a second seismic switch will be installed on the Class 1 piping connected to reactor coolant loop. The seismic switch will be responsive to frequencies from 0.1 Hz to 30 Hz. The switch on the basemat will be set at 0.1g corresponding to an OBE. The seismic switch is a triaxial low frequency acceleration sensor with adjustable threshold accelerations in three orthogonal directions. It operates with an internal rechargeable power supply. The minimum duration of the switch actuation is adjustable (6-20 seconds), and remains actuated as long as the setpoint is exceeded. Audio alarm will result once the seismic switch is actuated.

3.7.4.2.3 Triaxial Spectrum Recorder

The triaxial response spectrum recorder provides a permanent record of spectral accelerations at 12 discrete frequencies on all three axes. The recorded values in the main control room provides a basis to see whether the spectral acceleration levels at individual discrete frequencies are within, or above, the OBE response spectrum levels.

The response spectrum recorders will be responsive to a frequency range of 1 Hz to 30 Hz with appropriate damping value to facilitate comparison of spectral acceleration values associated with the OBE response spectra. They will be employed to provide more information on the seismic input and the potential plant seismic response property with no need to wait for detailed processing of the time-history accelerograph records.

A total of four response spectrum recorders are provided, one each at the following locations:

1. The top of the foundation base mat inside the reactor external building and near the vicinity of the time-history accelerograph of Subsection 3.7.4.2.1.

2. The class 1 piping connected to reactor coolant loop.

3. The concrete operating floor of the main control room.

4. The support to a Category I piping system.

In case the design site specific ground motion is 0.3g ZPA or higher, a fifth response spectrum recorder will be installed on the supporting pad of a Category I equipment structure.

3.7.4.2.4 Triaxial Peak Accelerograph

The peak accelerograph is a self contained passive device capable of permanently recording peak acceleration. It detects peak acceleration in a frequency range from 0.1 Hz to 20 Hz. Data from the peak accelerograph will be manually retrieved following an earthquake and will be used in the detailed evaluation of seismic performance of the plant structures, systems and components.

A total of three triaxial peak accelerographs are provided, one each at the following locations:

1. The reactor coolant pump motor

2. The Class 1 piping connected to reactor coolant loop.

3. The Category I piping outside the containment.

In case the design site specific ground motion is 0.3g ZPA or higher, a fourth peak accelerograph will be installed on the supporting pad of a Category I equipment structure.

3.7.4.2.5 Criteria for Instrument Location

The selection of the above locations for installing seismic instrument is based on the guidance provided in the USNRC Regulatory Guide 1.12, Revision 1, for an SSE acceleration of 0.3g or higher, unless as noted.

All instruments are accessible for inspection, test and service except for the instruments on the reactor coolant pump motor and the reactor coolant Class 1 piping which are accessible only during reactor shutdown.

Table 3.7-2 summarizes the locations of the seismic instruments.

3.7.4.2.6 Seismic Instrumentation Control Panel

An instrumentation panel located in the main control room will be provided to house the recording, playback and calibration units which are used in conjunction with the time-history accelerographs. It also contains the audio alarms and visual displays in association with the operation of the seismic switches and the response spectrum recorders.

3.7.4.3 Control Room Operator Notifications

Operator notification consists of alarms, indicating lights and graphical displays.

Audio and visual alarms will be provided in the main control room for the following parameters:

- o Containment foundation ZPA input in excess of 0.1g (OBE)
- o Actuation of any time-history accelerographs
- o Response spectral values in any frequency and any axis in excess of design OBE spectral accelerations as recorded by the response spectrum recorders.

The time-history accelerograph records will be played back to provide visual displays as needed after earthquakes.

3.7.4.4 Comparison of Measured and Predicted Responses

The plan for utilization of the seismic data includes both the function of the operator and engineering to evaluate the effects of an earthquake on the plant. For a detail description of the data flow, refer to Figure 3.7-20.

Initial determination of the earthquake effect is performed immediately after the earthquake by comparing the measured response spectra from the containment base mat with the OBE and SSE design response spectra for the corresponding location.

If the measured spectra exceed the OBE response spectra, the plant will be shutdown and a detailed analysis of the earthquake motion will be undertaken.

After an earthquake, the data from the seismic recording instruments are reviewed. See Figure 3.7-20. The data from these instruments will be analyzed to obtain the seismic accelerations experienced at the location of major Category I structures and equipment. The measured responses from the instruments will be used to evaluate seismic Category I structures and systems in which the spectra are compared with those used in the design to determine whether the OBE design level has been exceeded or not.

During shutdown as a result of OBE earthquake, the equipment mounted triaxial peak accelerographs will be used to determine if the design limitation of specific equipment to which it is fastened has been exceeded. If the measured responses are less than the values used in the design and qualification of the Seismic Category I structures, systems, and equipment and a visual inspection of the systems and components reveals no damage, the structure, system, or equipment is considered adequate for future operation. Otherwise, damage is corrected, and a new analysis is made to assure the adequacy of those items for future use.

3.7.4.5 Inservice Surveillance

Calibration and alignment on three orthogonal axes will be performed prior to fuel loading in order to assure proper operation. Periodic testing and calibration will be performed in accordance with technical specification.

3.7.5 References

1. "Use of Frequency Dependent Damping Values," Westinghouse letter NS-EPR-2955, Rahe, E.P. to J.P. Knight, dated August 16, 1984.

2. Gesinki, T.L., and Chiang, D., "Safety Analysis of the 17x17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident," WCAP-8236 (Proprietary), December 1973, and WCAP-8288 (Nonproprietary), January 1974.

3. Obermeyer, F.D., "Effective Structural Damping of the KEP L105 CRDM," WCAP-7427 (Proprietary), January 1970.

4. Obermeyer, F.D., WCAP-7427, Addendum 1 (Proprietary) December 1970.

5. "Standard for the Seismic Analysis of Safety-Related Nuclear Structures," ASCE Seismic Analysis Standard Committee, May, 1984, Draft.

6. Lin, C.W., "DEBLIN2 - A Computer Code to Synthesize Earthquake Acceleration Time Histories," WCAP-8867, November, 1976.

7 Lin, C.W., "Time History Input Development for the Seismic Analysis of Piping Systems," Journal of Pressure Vessel Technology, Vol. 102, No. 2, May 1980.

8. "Standard for the Seismic Analysis of Safety-Related Nuclear Structures," American Society of Civil Engineers, Draft, September 1983.

9. "Evaluation of Other Dynamic Loads and Load Combinations," Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, NUREG-1061, Vol. 4, December 1984.

10. Park, I.B., and Johnson, E.R., "Computationally Efficient Methods for Seismic Response from Flexible and Rigid Modes," Seismic and Dynamic Analysis Methods (C.W. Lin, ed.), ASME-BVP-81, New York, 1984.

11. Vashi, K.M., "Seismic Spectral Analysis for Structures Subjected To Non-Uniform Excitation," ASME Paper No., 83-PVP-69.

12. Lin, C.W., and Loceff, F., "A New Approach to Compute System Response with Multiple Support Response Spectra Input," Nuclear Engineering and Design 60, North-Holland Publishing Company, 1980, pp. 347-352.

13. Lin, C.W., and Guilinger, W.H., "System Response to Multiple Support Response Spectra Input," Proceedings of Sixth International SMIRT Conference, Paper K10/4, Paris, 1981.

TABLE 3.7-1

REGULATORY GUIDE 1.61 DAMPING VALUES FOR STRUCTURES OR COMPONENTS[(a)]

Structure of Component	Percent of Critical Damping Per Mode OBE	SSE
Welded steel structures	2	4
Bolted steel structures	4	7
Prestressed concrete structures	2	5
Reinforced concrete structures	4	7

a. Damping values for foundation material, used in foundation-structure interaction analysis, are not included in this table.

TABLE 3.7-2
SEISMIC MONITORING INSTRUMENTATION REQUIREMENTS

Instrumentation Location	Triaxial Time-History Accelerograph	Triaxial Response Spectrum Recorder	Triaxial Peak Accelerograph	Triaxial Seismic Switch
I. Free Field				
1. 500 ft from Reactor External Building	1*,#			
II. Inside Containment				
1. Basemat	1*	1*		1*
2. Operating Floor	1*			
3. Reactor Coolant Pump Motor			1	
4. Class 1 Piping		1	1	1*,x
III. Outside Containment				
1. Main Control Room	1*,x	1		
2. Cat. I Equip. Sup.		1x	1x	
3. Cat. I Piping Sup.		1		
4. Cat. I Piping			1	

* Readout and annunciated in the Main Control Room
\# May be omitted if site soil structure interaction is negligible
x May be omitted for design SSE less than 0.3g ZPA

(a,c

Figure 3.7-1 Safe Shutdown Earthquake Horizontal Response Spectra

(a,c

Figure 3.7-2 Safe Shutdown Earthquake Vertical Response Spectra

(a,c

Figure 3.7-3 Operating Basis Earthquake Horizontal Response Spectra

(a,c

Figure 3.7-4 Operating Basis Earthquake Vertical Response Spectra a,c

Figure 3.7-5 Comparison of Design Response Spectra
- Horizontal Direction 1

(a,c

Figure 3.7-6 Comparison of Design Response Spectra
- Horizontal Direction 2

(a,c

Figure 3.7-7 Comparison of Design Response Spectra - Vertical Direction

(a,c)

Figure 3.7-8 Westinghouse Position on Damping

(a,c)

Figure 3.7-9 Analytical Model - Seismic System

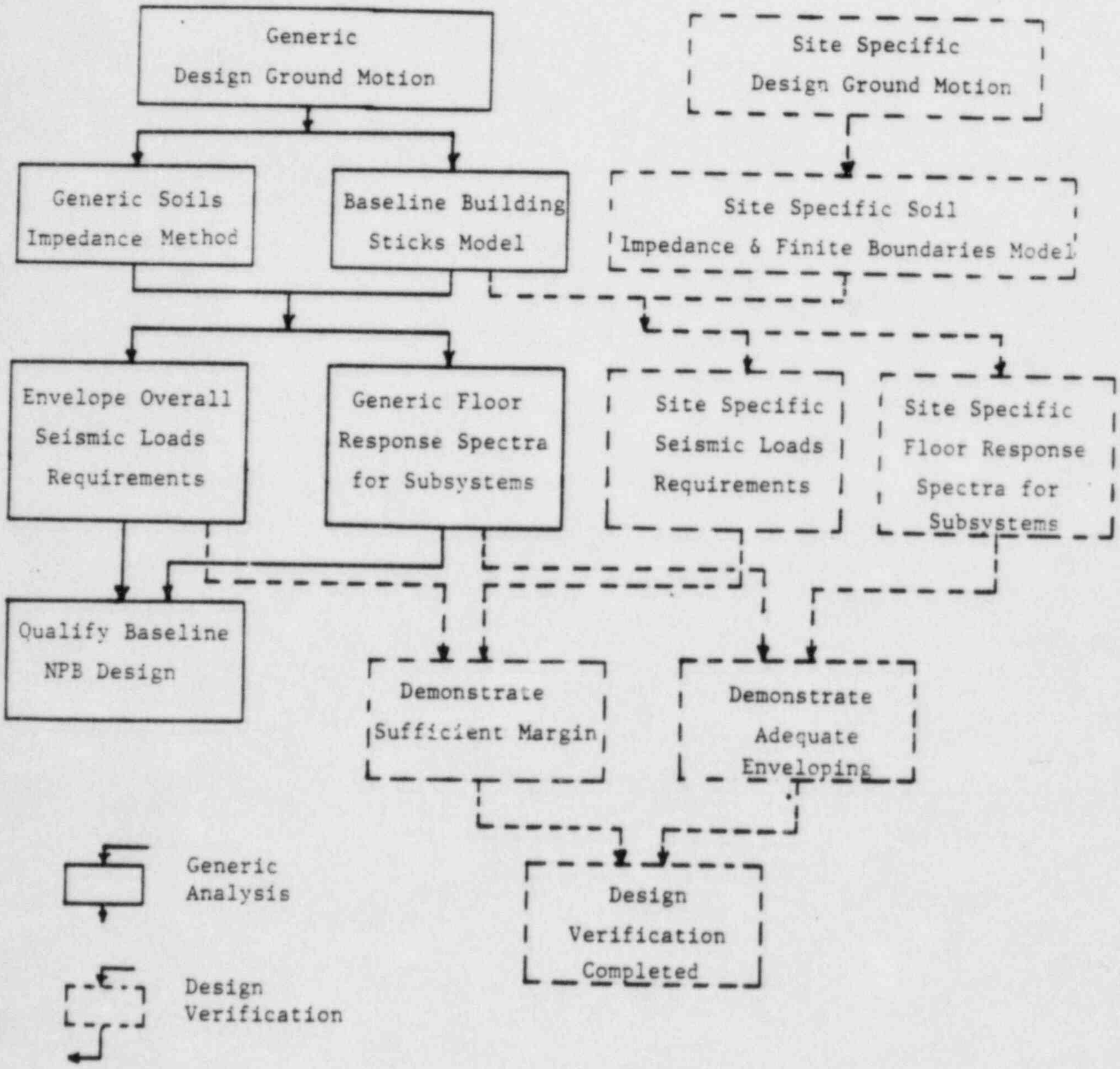


Figure 3.7-10 Seismic Analysis of Soil-Structure Systems

FIGURE 3.7-11 ENVELOPED FLOOR RESPONSE SPECTRA - OPERATING DECK(N-S)

(a,c)

(a,c)

FIGURE 3.7-12 ENVELOPED FLOOR RESPONSE SPECTRA - OPERATING DECK(E-W)

(a,c)

FIGURE 3.7-13 ENVELOPED FLOOR RESPONSE SPECTRA - OPERATING DECK(VERTICAL)

(a,c)

FIGURE 3.7-14 ENVELOPED FLOOR RESPONSE SPECTRA - RPV SUPPORT(N-S)

(a,c)

FIGURE 3.7-15 ENVELOPED FLOOR RESPONSE SPECTRA - RPV SUPPORT(E-W)

(a,c)

FIGURE 3.7-16 ENVELOPED FLOOR RESPONSE SPECTRA - RPV SUPPORT (VERTICAL)

FIGURE 3.7-17 ENVELOPED FLOOR RESPONSE SPECTRA - CONTROL ROOM(N-S)

(a,c)

FIGURE 3.7-18 ENVELCPED FLOOR RESPONSE SPECTRA - CONTROL ROOM(E-W)

(a,c)

(a,c)

FIGURE 3.7-19 ENVELOPED FLOOR RESPONSE SPECTRA - CONTROL ROOM(VERTICAL)

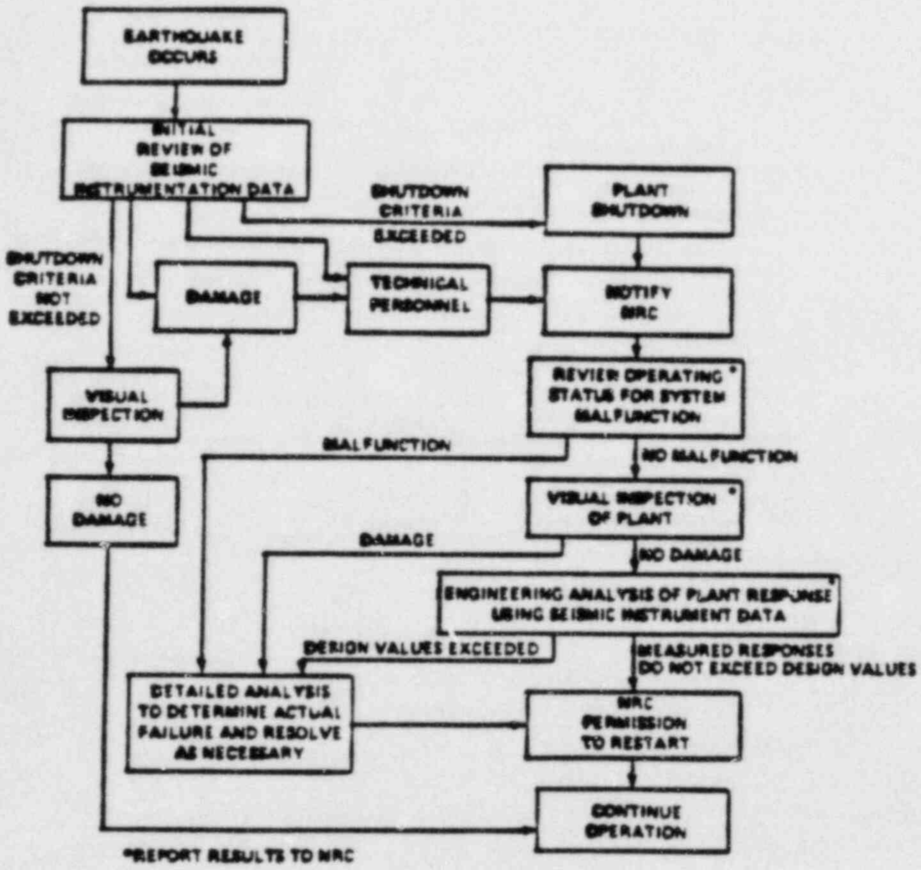


Figure 3.7-20 Post Seismic Event Data Utilization

3.8 DESIGN OF CATEGORY I STRUCTURES

A detailed description of the containment structures, internal structures and other Category I structures is provided in the following sections.

3.8.1 Concrete Containment

This section is not applicable to the Nuclear Power Block. The concrete shield building is included in Category I structures and is described in Subsection 3.8.4.

3.8.2 Steel Containment

3.8.2.1 Description of the Containment

The containment vessel is a free-standing, spherical, welded steel shell, 60 m inside diameter and 42 mm thick. The lower portion of the shell below elevation 92.2 m is encased between the building foundation concrete and the interior structure base concrete, without any structural connection between the steel and concrete. The strength provided by concrete encasement is ignored, and the shell thickness in the embedded segment is the same as in the upper portion. To reduce the secondary stresses in the shell in the area around elevation 92.2 m a strip of a compressible material will be provided all around the contact area.

The vessel includes the shell, equipment hatch, penetrations, airlocks, miscellaneous appurtenances and attachments. The containment penetrations, other than the equipment hatch and the airlocks, consist of the fuel transfer penetration, mechanical penetrations, and electrical penetrations. A fuel transfer tube is provided at elevation 88.7 m for transfer of fuel between the fuel pool and the containment refueling canal.

Mechanical penetrations are treated as fabricated piping assemblies meeting the requirements of ASME III, Subsection NE, and are assigned the same classification as the piping system that includes the assembly.

The process line and flued heads making up the pressure boundary will be consistent with the system piping materials; fabrication, inspection, and analysis requirements will be as required by ASME III, Subsection NE. All welds on the process pipe will be accessible for inspection in accordance with ASME Section XI.

Medium voltage electrical penetrations for reactor coolant pump power 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.

3.8.2.2 Applicable Codes, Standards, and Specifications

The steel containment is designed, fabricated and tested in accordance with the provisions of ASME Boiler and Pressure Vessel Code, Division 1, Section III, Subsection NE.

All structural steel non-pressure parts such as ladders, walkways, handrail, etc. will be designed in accordance with the American Institute of Steel Construction (AISC), "Specification for the Design, Fabrication and Erection of Structural Steel Buildings".

The containment penetrations, other than airlocks and the equipment hatch consist of electrical and piping penetrations. The portion of the penetrations consisting of the pipe sleeve welded to the vessel will be designed, fabricated, installed, and tested according to the requirements of the ASME Code, Section III, Subsection NE. The connections between the vessel pipe sleeve and the piping passing through the containment vessel shell will consist of a bellows assembly, flued head or other welded connection designed, fabricated, installed, and tested to meet the requirements of the particular system and Section III of the ASME code. The containment pipe sleeves in which electrical penetration assemblies are installed will be designed to meet the requirements of ASME Code, Section III as well as being compatible with the electrical penetration assemblies. The electrical penetration assemblies will be designed, fabricated, installed and tested in accordance with the requirements of IEEE 317-1976.

3.8.2.3 Loads and Load Combinations

The loads and the load combinations used in the design of the containment are in accordance with the requirements of ASME Code, Section III, Subsection NE.

Seismic loads are discussed in Section 3.7 of this module. Missile effects and pipe rupture loads are discussed in Sections 3.5 and 3.6, respectively, of this module. The design internal pressure is 46 psig. The design external pressure will be 2.0 psig. The containment interior structure is vented at the operating deck to allow LOCA pressure release to the upper containment. Therefore, the containment shell is not subjected to LOCA transient pressures. The load combinations are shown in Table 3.8-1.

3.8.2.4 Design and Analysis Procedures

The design and analysis procedures for the containment vessel conform to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Subsection NE. The following computer programs are employed for the 3-dimensional analysis of the containment shell.

1. "WECAN" - Westinghouse Electric Corporation Analysis Program.

2. "ASHSD" - "Dynamic Stress Analysis of Axisymmetric Structure Under Arbitrary Loading," developed by Ghosh and Wilson Program.

   "ASHSD" is used to analyse a two-dimensional model of the axisymmetric shell of revolution. The applied loads can vary in meridional and circumferential directions. The "WECAN" program is used to analyze a detailed three dimensional finite element model, and is used for areas where the vessel is not axisymmetric such as the regions around the penetrations.

Classical theory and analysis methods are also used for local areas such as the personnel locks and small penetrations. For conditions where compressive stresses occur, the critical buckling stress is checked against the provisions of code case N-284.

3.8.2.5 Structural Acceptance Criteria

The containment vessel will receive a code stamp. The vessel is designed and will be fabricated, installed and tested in accordance with the provisions of ASME Code, Section III, Subsection NE. The stress intensity limits for all load combinations are specified in Table 3.8-2. Critical buckling stresses are checked in accordance with the provisions of the ASME Code case N-284.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

Materials for the containment vessel, including equipment hatch, equipment and emergency personnel access locks, penetrations, attachments, and appurtenances will meet the requirements of NE-2000 of ASME-III. The basic containment material is SA-516, Grade 70 carbon steel. Stairways, ladders and platforms will be fabricated from A-36 carbon steel. Impact test requirements will be as specified in NE-2000. Charpy V-notch specimens and other details will be specified in accordance with the requirements of the ASME Code.

The quality control program involving welding procedures, erection tolerances, and nondestructive examination of both shop and field fabricated welds is in conformance with Articles NE-4000 and NE-5000 of the ASME Code. There are no special construction techniques used on the Class MC items.

3.8.2.7 Testing and Inservice Inspection Requirements

Testing of the Class MC items and the pipe assemblies forming the pressure boundary within the Containment Vessel will be in accordance with the provisions of articles NE-6000 and NC-6000 of the ASME code respectively. Periodic testing of these items will also be done as required by 10 CFR 50, Appendix J.

3.8.3 Concrete and Steel Internal Structures

3.8.3.1 Description of the Internal Structures

The concrete and steel internal structures support the reactor coolant system (RCS) components and related piping systems and equipment inside the containment pressure boundary. The concrete structures also provide radiation shielding. The internal structures consist of the primary shield wall, various compartment walls, refueling canal walls, operating floor, and intermediate slabs and platforms.

A description of the main structures that constitute internal structures is given in the following paragraphs; their locations are shown in Figure 1.2-2 of RESAR-SP/90 PDA Module 3, "Introduction and and Site". The primary shield wall provides radiation shielding, structural support for reactor vessel nozzles and also protection from internal missiles. It is designed to withstand post-LOCA pressures and temperatures, and reactions from reactor vessel supports. The primary shield is six feet thick. The reactor pressure vessel (RPV) support system consists of four seats under two hot leg and two cold leg nozzles, which are spaced 90° apart in the primary shield wall. Under these seats, steel weldments embedded in the primary shield are provided to transfer reactor loads to the primary shield wall.

The steam generators (SG) are supported vertically by four steel columns, bolted to support pads on the vessel and basemat embedments. Steel framing attached to the compartment walls provides SG lateral support.

For the reactor coolant pumps, three steel columns, bolted to the pump pads and basemat embedments, provide vertical support, and steel tie rods anchored to the primary shield and other concrete partition walls provide the lateral restraint.

In each loop, major equipment is enclosed in three compartments. Two compartments enclose the steam generators and corresponding reactor coolant pumps. The third compartment encloses the reactor vessel and the incore instrumentation assemblies. The walls of these compartments are at least 3 feet thick and are designed to withstand LOCA pressures and to provide radiation shielding.

The refueling canal is a reinforced concrete structure provided for the underwater transfer of fuel assemblies and for the storage of the reactor internals. The entire refueling canal is lined with stainless steel plate. The fuel transfer tube connects the refueling canal to the spent fuel pool. During refueling operations, the canal is filled with borated water to a depth that limits the radiation from fuel assemblies to acceptable levels.

The operating floor at elevation 100m provides access for operating personnel and for associated operating functions. It is a 0.75 meter thick reinforced concrete slab with openings for venting the equipment compartments below. The floor slab is supported by the refueling canal walls, equipment compartment walls and vertical steel columns spaced 3m to 6m apart.

The hatch covers and other removable structures are also included in the category of internal structures. The removable slabs and hatch covers are tied down to eliminate any possibility of these becoming missiles during an accident.

3.8.3.2 Applicable Codes, Standards, and Specifications

The following codes and standards are applicable to the design, fabrication, and testing of the internal structures.

1. American Institute of Steel Construction, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings."

2. American Concrete Institute, "Code Requirements for Nuclear Safety-Related Structures," ACI 349.

3. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF.

3.8.3.3 Loads and Load Combinations

The loads and load combinations are the same as for other Category 1 structures described in Subsection 3.8.4.3 and the associated tables, except that wind loads (W) and tornado loads (W_t) are not applicable to the design of the internal structures because of the protection provided by the containment shell. These loading terms are excluded in the load combinations for the internal structure.

3.8.3.4 Design and Analysis Procedures

The layout of internal structures, as shown in general arrangement drawings (Figure 1.2-2, sheets 1 through 9 of RESAR-SP/90 PDA Module 3, "Introduction and Site") is similar to the internal structures in other PWR containments. Preliminary analyses indicate the adequacy of the internal structures. For the analysis of complex structures, finite element analysis programs such as WECAN are used.

3.8.3.5 Structural Acceptance Criteria

The concrete internal structures are designed in accordance with the requirements of ACI-349 (Reference 2 of Subsection 3.8.3.2), using the strength design method. The design of steel structures, excluding the equipment supports, follows the AISC specification (Reference 1 of Subsection 3.8.3.2). The RCS equipment supports are designed in accordance with ASME Subsection NF. (Reference 3 of Subsection 3.8.3.2)

3.8.3.6 Materials

A description of major materials is given in Subsection 3.8.4.6. The compressive strength of concrete used for Category I internal structures, $f_c^{'}$ is 4000 psi.

3.8.3.7 Testing and Inservice Inspection Requirements

No testing required except Quality Control.

3.8.4 Other Seismic Category I Structures

The major Category I structure covered by this section is the Reactor External Building. This includes the Shield Building, the Auxiliary Equipment Area, the Fuel Handling Area, the Control Complex Area, the Diesel Generator Building, the Main Steam Tunnel, and the Essential Safety Facility Area.

3.8.4.1 Description of the Structures

The containment shield building consists of a reinforced concrete cylinder and a hemispherical dome supported on a flat circular concrete basemat. The inside radius of the cylindrical and the spherical segments is 32m. The thickness of the dome is 0.5m, and the thickness of the cylindrical wall is 0.9m. The shield building houses the steel containment vessel, and is designed to provide radiation shielding as well as missile protection for the steel containment and other safety related structures. An outline of the shield building is shown in Figure 1.2-2, sheets 1 through 9 of RESAR-SP/90 PDA Module 3, "Introduction and Site".

An annular space is provided between the containment vessel and containment shield building above elevation 92.2 m. The annular space provides a controlled air volume for filtering and access to penetrations for testing and inspection.

The portion of the reactor building below elevation 92.2 m is an extension of the auxiliary building, and is used to house safety related equipment. The auxiliary building houses the chemical and volume control system (CVCS), emergency core cooling system (ECCS), residual heat removal system (RHR), heating ventilation and air-conditioning (HVAC, systems, and other equipment. It is a reinforced concrete structure composed of a foundation basemat, walls, columns, beams, and floor slabs.

The fuel handling area includes the spent fuel storage pool and the new fuel storage area. The spent fuel pool is a concrete box, with a stainless steel lining on the inside surface. The new fuel storage area is a separate reinforced concrete pit providing temporary dry storage for the new fuel assemblies. The transfer tube transfers fuel assemblies between the fuel handling building and the containment.

The control complex area is part of the Nuclear Power Block and is located between the containment and the turbine building. The area includes the control room, cable spreading rooms, switchgear, and HVAC equipment and a few nonsafety related components. The diesel-generators are located adjacent to the reactor building with floors at elevation 92.2 m and elevation 100.0 m. These are reinforced concrete structures and house diesel-generators and cranes for equipment handling. Air handling, exhaust and silencing equipment is also installed in this area. Figure 1.2-2, sheets 1 through 9 of RESAR-SP/90 PDA Module 3, "Introduction and Site" shows the layout and the general arrangement of these structures.

3.8.4.2 Applicable Codes, Standards, and Specifications

Category I structures will be designed in accordance with the codes and standards listed below:

1. American Institute of Steel Construction, "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings".

2. American Concrete Institute, "Code Requirements for Nuclear Safety-Related Structures", ACI-349.

3.8.4.3 Loads and Load Combinations

3.8.4.3.1 Loads

The following loads are considered in design and evaluation of Category I structures.

3.8.4.3.1.1 Normal Loads

Dead Loads (D)

The dead loads include the weight of framing, roofs, floors, walls, partitions, platforms, shielding, earthfill, and all permanently attached equipment, piping conduits and cable trays. The vertical and lateral pressures of liquids are also treated as dead loads, as provided in Subsection 9.3.3 of ACI 349.

Live Loads (L)

Live loads include all loadings superimposed by the use and occupancy of the building and not permanently fixed to the structure.

Following are the minimum live loads for use in the design

	kg/m^2	lb/ft^2
o Stairs and walkways - distributed load - or a moving concentrated load of 454 kg (1,000 pounds)	488	100
o Railings, 30 kg/m or 91 kg (20 lb /lineal ft. or 200 lb) applied in any direction at top of railing	-	-
o Platforms and gratings	488	100
o Ground floors	1220	250
o Engineered safeguards area	976	200
o Elevated floors not specified otherwise	976	200

No live load reduction is allowed for buildings of the Nuclear Power Block. Impact and the effect of concentrated loads is considered. Temporary loads which may exist during construction are considered if they exceed the floor design load. A minimum construction load of 50 psf is used on all structures. Other loads which vary with intensity and occurrence, such as soil pressure (H), are treated as live load.

Operating Basis Pipe and Equipment Loads (R_o)

All structural members subjected to pipe loads are designed for these loads for normal operating or shutdown conditions, based on the most critical transient or steady-state conditions.

All structural members are designed for the maximum weight of all equipment which they support.

In the design of the operating deck, refueling canal, steam generator enclosures, etc., impact loads associated with handling, refueling, and maintenance is considered, so that functional integrity of safety equipment and structures is not impaired.

Operating Basis Temperature Loads (T_o)

The location and magnitude of any temperature changes and gradients affecting the structure is investigated and incorporated in the design.

3.8.4.3.1.2 Environmental Loads

Wind (W) and Tornado Loads (W_t)

All structures exposed to outside environment are designed to resist wind and tornado loads, as specified in Section 3.3 of this module. In addition, all Seismic Category I structures, unless otherwise protected, and structures protecting Seismic Category I systems, components or equipment, are designed to remain functional when subjected to wind generated missiles. Seismic Category I structures may sustain local missile damage such as partial penetration and local cracking and/or permanent deformation, provided that structural integrity is maintained, perforation of air controlled environments is precluded and Seismic Category I systems, components and equipment are not subject to damage by secondary missiles, such as from concrete spalling.

Seismic Loads - Operating Basis Earthquake (E_o) and Safe Shutdown Earthquake (E_s)

Category I structures are designed for the Operating Basis and Safe Shutdown Earthquake as defined in Section 3.7 of this module.

3.8.4.3.1.3 Design Basis Accident Loads

These loads consist of the following effects resulting from a loss-of-coolant, accident or other pipe ruptures.

Pressure Load (P_a)

There are two distinct types of pressures associated with a design basis accident. One is the short duration differential transient pressures existing in the various compartments during initial energy release. The other is long term pressure after the accident. Both pressure types are considered in the design.

Jet Impingement Load (Y_j)

This is the dynamic impact load resulting from pressurized fluid jet being ejected from a postulated pipe rupture.

Missile Load (Y_m)

This is the load due to a postulated missile (including pipe whip) associated with a design basis accident.

The operating deck and upper reactor vessel internal storage area in the refueling canal are evaluated for reactor vessel head drop.

Pipe and Equipment Supports Thermal Reaction Loads (R_a)

Pipe and equipment supports reaction load under thermal conditions generated by the postulated event and including R_o.

Thermal Loads (T_a)

These are the overall and local thermal loads associated with a design basis accident.

Dynamic Pipe Break Reaction Load (Y_r)

This is the dynamic reaction loads on pipe and equipment restraints or support foundations due to a pipe break.

3.8.4.3.2 Load Combinations

The load combinations and allowable stresses used in the design of Category I structures other than the containment vessel are given in Tables 3.8-3 and 3.8-4.

3.8.4.4 Design and Analysis Procedures

The shield building and its attachments are designed in accordance with ACI-349. The reinforced concrete foundation mat, which is constructed as an integral part of the shield building, and the other Category I structures are also within the jurisdiction of the above mentioned code. The load combinations and allowable stresses used in the design are given in Tables 3.8-3 and [illegible]. The structural analysis of Category I structures is done using the finite element analysis computer code, "WECAN". The building model is assumed supported on linear soil springs which simulate the foundation conditions. The total structure is also checked for overturning and sliding due to lateral loads.

3.8.4.5 Structural Acceptance Criteria

All Category I structures are designed in accordance with the provisions of ACI-349, as modified by Regulatory Guide 1.142, using the strength design method. The allowable stresses in steel elements are governed by the provisions of the AISC Code.

3.8.4.6 Materials

Structural Steel

1. Plates, decking, and their connections, stairs and grating - ASTM A36 unless otherwise noted.

2. Shapes, base plates, and their connection materials - ASTM A572 Grade 50 or ASTM A36 or ASTM A516 Grade 60 or 70.

3. Bolts - ASTM A307 and ASTM A325, or ASTM A490 when necessary, or ASTM A193 Grade B7 or B8 Class 2, ASTM A194 Grade 4, 7, or 8 and ASTM A320 Grand L7.

4. Consumables (Electrodes, Fluxes, Bare Electrodes, etc.) - Metal deposited to have a minimum f_y = 50,000 psi and a minimum elongation of 20 percent in 2 inches.

5. Pipe - ASTM A106 Grade B or C, ASTM A333 Grade 1, or ASTM A155 Grade KCF 60 or 70.

6. Stainless Steel Plate - ASTM A240 Type 304.

7. Forgings - Carbon Steel ASTM A350 Grade LF1 for welding.

8. Castings - Carbon Steel ASTM A216 Grade WCB, or A352 Grade LCB.

Concrete and Embedded Items

1. Concrete - compressive strength of concrete, f'_c is 4000 psi.

2. Reinforcing - ASTM A615 Grade 60.

3. Embedded Steel - ASTM A36 or ASTM A572 Grade 50.

4. Welded Wire Fabric - ASTM A497.

5. Mechanical Rebar Connector - Cadweld sleeve ASTM A519 or equivalent with tensile strength not less than 125 percent of minimum yield strength of reinforcing bar.

6. Studs - Mild Steel tensile strength not less than 55,000 psi and 20 percent minimum elongation in 2 inches.

1. Grating, Ladders, Handrails	Galvanized steel (require special approval for use inside containment.)
2. Painting Materials	Inorganic Zinc Silicate, Epoxy Polyamide, Alkyl Silicate Inorganic Zinc.

Unacceptable Materials

The following materials shall not be used unless specifically approved for each particular application.

- Low-alloy Steel
- Aluminum
- Flammable materials (e.g., certain foam-plastic insulations) unless completely enclosed in steel or other fire-proof enclosures, and specifically approved in each particular case.

3.8.4.7 Testing and Inservice Surveillance Requirements

There will be no testing or inservice surveillance for the shield building and other Category I structures, beyond the quality control tests performed during the construction.

3.8.5 Foundations

3.8.5.1 Description of the Foundations

The Reactor External Building foundation is a reinforced concrete mat supported directly on firm soil or sound rock. (Refer to WAPWR plant layout Figures 1.2-2, Sheets 8 and 9, of RESAR-SP/90 PDA Module 3, "Introduction and Site"). The Reactor External Building foundation is separated from the foundation mats of other adjacent structures to eliminate any structural interaction. The thickness of the foundation will be determined from site - specific analyses which consider the specific soil conditions.

3.8.5.2 Applicable Codes, Standards, and Specifications

Reinforced concrete foundations are designed as Category I structures as described in Section 3.8.4.

3.8.5.3 Loads and Loading Combinations

The loads and load combinations are the same as for other Category I concrete structures. (Refer to Table 3.8-3 for load combinations and load factors for Category I concrete structures).

3.8.5.4 Design and Analysis Procedures

The reinforced concrete foundations of Category I structures are analyzed by linear elastic methods 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 behaves as a flat plate on an elastic subgrade. The vertical loads are transferred to the foundation medium by bearing. The horizontal loads are transferred to the foundation medium by friction and/or shear keys.

3.8.5.5 Structural Acceptance Criteria

The reinforced concrete foundation mat is designed in accordance with the provisions of ACI-349, Code for concrete structures. Factors of safety against overturning, sliding and floatation are defined below.

For Combination	Minimum Factors of Safety		
	Overturning	Sliding	Floatation
a. D + H + E	1.5	1.5	--
b. D + H + W	1.5	1.5	--
c. D + H + E'	1.1	1.1	--
d. D + H + W_t	1.1	1.1	--
e. D + F'	--	--	1.1

Where: D - Dead loads or their related internal moments and forces, including any permanent equipment 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. Live load is included as applicable in any particular situation.

H - is the lateral earth pressure.

F' - is the bouyant force of the design basis flood.

Severe environmental loads include:

E - Loads generated by the operating basis earthquake.

W - Loads generated by the design wind specified for the plant.

Extreme environmental loads include:

E' - Loads generated by the safe shutdown earthquake.

W_t - Loads generated by the design tornado specified for the plant.

3.8.5.6 Materials

Materials are described in Subsection 3.8.4.6.

3.8.5.7 Testing and Inservice Inspection Requirements

No testing required except normal Quality Control during construction.

TABLE 3.8-1

CONTAINMENT LOAD COMBINATIONS AND LOAD FACTORS

Category	D	L	P_t	P_a	T_t	T_a	E_o	E_s	R_a	R_r
Test	1.0	1.0	1.0	-	1.0	-	-	-	-	-
Design	1.0	1.0	-	1.0	-	1.0	-	-	1.0	-
Level 'A' Service Limit	1.0	1.0	-	1.0	-	1.0	-	-	1.0	-
Level 'B' Service Limit	1.0	1.0	-	1.0	-	1.0	1.0	-	1.0	-
Level 'C' Service Limit	1.0	1.0	-	1.0	-	1.0	-	1.0	1.0	-
Level 'D' Service Limit	1.0	1.0	-	1.0	-	1.0	-	1.0	1.0	1.0

Symbols:

D = dead loads.
E_o = operating basis earthquake.
E_s = safe shutdown earthquake.
L = live load.
P_t = test pressure load.
P_a = accident/incident maximum pressure.
R_a = piping loads.
T_a = accident/incident temperature thermal load.
T_t = test temperature thermal load.
R_r = loads due to pipe rupture.

TABLE 3.8-2

STRESS INTENSITY LIMITS FOR STEEL CONTAINMENT

Load Categories		Primary Stresses: Gen. Mem. P_m	Primary Stresses: Local Mem. P_L	Primary Stresses: Bending & Local Mem. $P_b + P_L$ (6)	Primary & Secondary Stresses $P_L + P_b + Q$	Peak Stresses $P_L + P_b + Q + F$
Testing Condition	Pneumatic	$0.75S_y$	$1.15S_y$	$1.15S_y$	N/A(2)	fatigue evaluation(5)
Design Condition		$1.0S_{mc}$	$1.5S_{mc}$	$1.5S_{mc}$	N/A	N/A
Level A Service Limit(1)		$1.0S_{mc}$	$1.5S_{mc}$	$1.5S_{mc}$	$3.0S_{mi}$	fatigue evaluation
Level B Service Limit		$1.0S_{mc}$	$1.5S_{mc}$	$1.5S_{mc}$	$3.0S_{mi}$	fatigue evaluation
Level C Service Limit	Not Integral and Continuous	$1.0S_{mc}$	$1.5S_{mc}$	$1.5S_{mc}$	$3.0S_{mi}$	N/A
	Integral and Continuous(4),(7)	$1.2S_{mc}$ or * $1.0S_y$	$1.8S_{mc}$ or * $1.5S_y$	$1.8S_{mc}$ or * $1.5S_y$	N/A	N/A
Level D Service Limit	Not Integral and Continuous(4)	$1.2S_{mc}$ or * $1.0S_y$	$1.8S_{mc}$ or * $1.5S_y$	$1.8S_{mc}$ or * $1.5S_y$	N/A	N/A
	Integ. & Con. — Elas. Analysis(3)	S_f	$1.5S_f$	$1.5S_f$	N/A	N/A
	Integ. & Con. — Inelas. Anlys.(3)	S_f	S_f	S_f		
Post-Flooding Condition(4)		$1.2S_{mc}$ or * $1.0S_y$	$1.8S_{mc}$ or * $1.5S_y$	$1.8S_{mc}$ or * $1.5S_y$	$3S_{mi}$	N/A(2)

NOTES:
(1) The allowable stress intensity S_{mi} shall be the S_m listed in Table 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 Table I-1.0.
(4) These limits identified by (*) sign indicate a choice of the larger of 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 also to the partial penetration welds.
(8) Values shown are applicable when $P_L \leq 0.67S_y$. 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 $[2.5 - 1.5 (P_L/S_y)]\ S_y$.

TABLE 3.8-3

LOAD COMBINATIONS AND LOAD FACTORS FOR CATEGORY I CONCRETE STRUCTURES

Combination #		Load Combinations and Factors										
		1	2	3	4	5	6	7	8	9	10	11
Load Description												
Dead	D	1.4	1.4	1.4	1.0	1.0	1.0	1.0	1.0	1.05	1.05	1.05
Liquid	F	1.4	1.4	1.4	1.0	1.0	1.0	1.0	1.0	1.05	1.05	1.05
Live	L	1.7	1.7	1.7	1.0	1.0	1.0	1.0	1.0	1.3	1.3	1.3
Earth	H	1.7	1.7	1.7	1.0	1.0	1.0	1.0	1.0	1.3	1.3	1.3
Normal reaction	Ro	1.7	1.7	1.7	1.0	1.0				1.3	1.3	1.3
Normal thermal	To				1.0	1.0				1.4	1.4	1.4
OBE	Eo		1.9					1.25(3)			1.3	
Wind	W			1.7								1.3
SSE	Es				1.0				1.0(3)			
Tornado	Wt					1.0						
Accident thermal	Ta						1.0	1.0	1.0			
Accident thermal reactions	Ra						1.0	1.0	1.0			
Accident pressure	Pa						1.5	1.15	1.0			
Accident jet, Missile reactions	Y							1.0	1.0			

Notes:

1) Design per ACI-349 Strength Design Method for all load combinations
2) Where any load reduces the effects of other loads, the corresponding coefficient for that load shall be taken as 0.9 if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise the coefficient for the load shall be taken as zero.
3) Seismic loads will only be combined with ruptures of pipes that are not seismically supported.

TABLE 3.8-4

LOAD COMBINATIONS AND LOAD FACTORS FOR CATEGORY I STEEL STRUCTURES

Combination #		Load Combinations and Factors										
		1	2	3	4	5	6	7	8	9	10	11
Load Description												
Dead	D	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Liquid	F	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Live	L	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Earth	H	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Normal reaction	Ro	1.0	1.0	1.0	1.0	1.0				1.0	1.0	1.0
Normal thermal	To				1.0	1.0				1.0	1.0	1.0
OBE	Eo		1.0					1.0(3)			1.0	
Wind	W			1.0								1.0
SSE	Es				1.0				1.0(3)			
Tornado	Wt					1.0						
Accident thermal	Ta						1.0	1.0	1.0			
Accident thermal reactions	Ra						1.0	1.0	1.0			
Accident pressure	Pa						1.0	1.0	1.0			
Accident jet, Missile reactions	Y							1.0	1.0			

Allowable Stress: 1.0S (combinations 1–5), 1.6S (combinations 6–8), 1.5S (combinations 9–11)

Notes:
1) S denotes allowable stresses per AISC Specification, Part I
2) Where any load reduces the effects of other loads the corresponding coefficient for that load shall be taken as zero unless it can be demonstrated that the load is always present or occurs simultaneously with the other loads.
3) Seismic loads will only be combined with ruptures of pipes that are not seismically supported.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

The following five operating conditions, as defined in Section III of the ASME Code, are considered in the design of the class 1 components, component supports, and reactor internals.

A. Level A Service Conditions (Normal Conditions)

Any condition in the course of start-up, operation in the design power range, hot standby and system shutdown, other than Level B, Level C, Level D Service conditions or testing conditions.

B. Level B Service Conditions (Upset Conditions - Incidents of Moderate Frequency)

Any deviations from Level A Service Conditions anticipated to occur often enough that design should include a capability to withstand the service conditions without operational impairment. The Level B Service Conditions include those transients which result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of load or power. Level B Service Conditions include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of a Level B Service Condition shall be included in the design specifications.

C. Level C Service Conditions (Emergency Conditions - Infrequent Incidents)

Those deviations from Level A Service Conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events shall not cause more than 25 stress cycles having an S_a value greater than that for 10^6 cycles from the applicable fatigue design curves of the ASME Code Section III.

D. Level D Service Conditions (Faulted Conditions - Limiting Faults)

Those combinations of service conditions associated with extremely low probability, postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that consideration of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

E. Testing Conditions

Testing conditions are those pressure overload tests including hydrostatic tests, pneumatic tests, and leak tests specified. Other types of tests shall be classified under normal conditions.

To provide the necessary high degree of integrity for the class 1 equipment, the transients selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. To a large extent, these transients are based on engineering judgment and experience and are considered to be of such magnitude and/or frequency to be significant in the component design and fatigue evaluation processes. Pertinent variations in pressure, fluid temperature, and flow are used to describe these transients.

The design transients and the number of cycles that are normally used for fatigue evaluation of major Reactor Coolant System components are summarized in Table 3.9-1. In accordance with ASME III, Level C and Level D Service

Conditions are not included in fatigue evaluations. Also, class 2 and class 3 piping systems do not require thermal transient analysis.

3.9.1.1.1 Level A Service Conditions (Normal Conditions)

The following primary system transients are considered Level A Service Conditions:

A. Reactor coolant pump start-up/shutdown
B. Plant heatup and cooldown
C. Unit loading and unloading between 0 and 15 percent of full power
D. Unit loading and unloading between 15 and 100 percent
E. Reduced temperature return to power
F. Step load increase and decrease of 10 percent of full power
G. Large step load decrease with steam dump
H. Load regulation
I. Boron concentration equalization
J. Feedwater cycling
K. Loop out of service
L. Refueling
M. Turbine roll test
N. Primary side leakage test
O. Secondary side leakage test
P. Core lifetime extension
Q. Feedwater heaters out of service

A. Reactor Coolant Pumps (RCPs) Start-up and Shutdown

The RCPs are started and stopped during routine operations such as RCS venting, plant heatup and cooldown, and in connection with recovery from certain transients such as loop out of service and loss of power. Other (undefined) circumstances may also require pump starting and stopping.

Of the spectrum of RCS pressure and temperature conditions under which these operations may occur, three conditions have been selected for defining transients:

a. Cold condition - 70°F and 400 psig[a]

b. Pump restart condition[b] - 100°F and 400 psig

c. Hot condition - 557°F and 2235 psig

NOTE: These pressure and temperature values are defined for use in the design and fatigue evaluation processes. Actual pump starting and stopping conditions may be controlled by other factors such as reactor vessel material ductility considerations.

For RCP starting and stopping operations, it is assumed that variations in RCS primary side temperature, and in pressurizer pressure and temperature are negligible, and that the steam generator secondary side is completely unaffected. The only significant variables are the primary system flow and the pressure changes resulting from the pump operations.

The following cases are considered:

Case 1 - First Pump Start-up (Last Pump Shutdown)

Variations in reactor coolant loop flow accompany start-up of the first pump, both in the loop containing the first pump pump being started and in the other loops (loops in which the pumps remain idle but reverse flow is developed). This case involves a higher dynamic pressure loss in the loop containing the pump being started, but the magnitude of the flow change is less than in Case 2. For the last pump shutdown case, the transient is the reverse of the first pump start-up transient.

(a) The lowest pressure required for RCP operation may be 300 psig. However, 400 psig is considered a conservative value for design purposes.

(b) These conditions are included to take care of situations requiring stopping and restarting the pumps after plant heatup has commenced.

Case 2 - Last Pump Start-up (First Pump Shutdown)

This case conservatively represents the variations in reactor coolant loop flow accompanying start-up of the second and third pumps. Initially flow exists through the second and third loops in the reverse direction as the result of starting the first pump. The remaining pumps in these loops are then started in sequence and a new equilibrium flow is established. The magnitude of flow reversal is the largest in the loop containing the last pump to be started. For the first pump shutdown case, the transient is the reverse of the last pump start-up transient.

Table 3.9-10 includes the RCP start-ups and shutdowns associated with RCS heatup and cooldown.

The values shown in this table represent the design conditions for the pump starting and stopping operations. The processes by which these conditions are attained are parts of other operations and are not defined here. For example, the RCS venting operation involves pressurizing the system to approximately 400 psig with a charging pump, starting and stopping one RCP to purge out air during the venting, then depressurizing back to essentially atmospheric pressure. For design purposes this process is assumed to be repeated four times per loop for each of 200 venting operations during the plant lifetime. This establishes the design value of 800 starts/stops for each RCP associated with the venting operation.

Another consideration is that the loop flow change associated with pump start-up develops a pressure differential in the normal (forward) direction across the divider plates of the steam generator in that loop. In the loops undergoing reverse flow, the direction of the divider plate ΔP is reversed. The magnitudes of the dynamic pressure drops depend on the volumetric flow rate through the loop and on the density and viscosity of the reactor coolant.

B. Plant Heatup and Cooldown

The plant heatup and cooldown operations are conservatively represented by uniform temperature ramps of 100°F per hour when the system temperature is above 350°F. This rate bounds both potential nuclear heatup operations and cooldown using the steam dump system.

Below 350°F, only reactor coolant pump heat and small amounts of decay heat are available to heat the RCS. Cooldown between 350°F and the shutdown temperature of 120°F[1] is accomplished by the Residual Heat Removal System (RHRS). In this range, a uniform ramp rate of 50°F per hour bounds the temperature rate change resulting from these operations.

Rates in excess of the above values will not be attained in actual practice because of other limitations such as:

a. material ductility considerations which establish maximum permissible temperature rates of change, as functions of RCS pressure and temperature.

b. Reduction in heatup rates on pump energy only because of increasing losses and decreasing pump power as system temperature increases.

c. Reduction in cooldown rates as steam dump and residual heat removal approach their respective temperature endpoints.

d. Interruptions in the heatup and cooldown cycles due to such factors as protection against RCS cold overpressure, pressurizer steam bubble formation, control rod withdrawal, sampling, water chemistry control and gas adjustments.

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1. Reactor Coolant System temperature can be as low as 70°F during the shutdown period. Between 70°F and 120°F the temperature is assumed to change very slowly, without causing any significant thermal transient effects.

The number of such complete heatup and cooldown operations is specified as 200 each, which corresponds to five such occurrences per year for the 40-year plant design life.

The plant design permits plant heatups and cooldowns to be conducted in accordance with either of two basic modes:

o The "steam bubble" mode, which involves maintenance of a steam cushion in the pressurizer to the maximum extent possible during plant heatup and cooldown. This helps to protect against reactor vessel overpressurization at low reactor coolant temperatures.

o The "water solid" or "conventional" mode, which permits early reactor coolant pump operation during plant heatup, resulting in more rapid plant heatup and earlier attainment of the no-load temperature.

C. Unit Loading and Unloading Between 0 and 15 Percent of Full Power

The unit loading and unloading cases between 0 and 15 percent power are represented by continuous and uniform ramp power changes, requiring 30 minutes for loading and 5 minutes for unloading. During loading, reactor coolant temperatures are increased from the no-load value to the normal load program temperatures at the 15 percent power level. The reverse temperature change occurs during unloading.

Prior to loading, it is assumed that the plant is at hot standby under feedwater cycling conditions. Loading commences and the feedwater temperature increases from the no-load value to the 15 percent power value controlled by steam dump and turbine start-up heat input to the feedwater. Prior to unloading, feedwater heating is terminated, steam dump is reduced to residual heat removal requirements, and feedwater temperature decreases from the 15 percent power value to the no-load value. The RCS pressure and pressurizer pressure are assumed to remain constant at the normal operating value during these operations.

The number of these loading and unloading transients is assumed to be 500 each during the 40-year plant design life.

D. Unit Loading and Unloading Between 15 and 100 Percent of Full Power Per Minute

The unit loading and unloading operations are conservatively represented by continuous and uniform ramp power change of 5 percent per minute between 15 percent and 100% power levels. This load swing is the maximum possible consistent with operation under automatic reactor control. It should be noted that in actual practice, changes in power level may take place at rates less than 5 percent per minute. The reactor temperature will vary with load as prescribed by the reactor control system.

The number of loading and unloading operations is specified as 13,200. One loading operation per day yields 14,600 such operations during the 40-year design life of the plant. By assuming a 90 percent availability factor, this number is reduced to 13,200.

It is also possible that as many as 2000 of the loading operations may be conducted in accordance with the "reduced temperature return to power" transient discussed in the following section. Both of these transients are evaluated to determine which is the more severe for the particular component design. If the reduced temperature mode is more severe, then 2000 occurrences of that transient should replace 2000 occurrences of the 5 percent per minute loading operation, reducing the number of 5 percent per minute loadings to 11,200.

E. Reduced Temperature Return to Power

The reduced temperature return to power operation is designed to improve the spinning reserve capabilities of the plant during load follow operations. The transient will normally begin at the ebb (50 percent) of a load follow cycle and will proceed at a rapid positive rate (typically 5 percent per minute) until the abilities of the control rods and the coolant temperature reduction (negative moderator coefficient) to supply reactivity are exhausted. At that point further power increases are limited by the ability of the boron system to dilute the reactor coolant. The reduction in primary coolant temperature is limited by the protection system to about 20°F below the programmed value.

The reduced temperature return to power operation is not intended for daily use. It is designed to supply additional plant capabilities when required because of network fault or upset condition. Hence, this mode of operation is not expected to be used more than once a week in practice (2000 times in 40 years).

If for a particular component this transient is more severe than the 5 percent per minute loading described in the preceding section, then 2000 occurrences of the 5 percent per minute loading for that component should be replaced by the reduced temperature return to power transient.

F. Step Load Increase and Decrease of 10 Percent of Full Power

The 10 percent step change in load demand results from disturbances in the electrical network into which the plant output is tied. The reactor control system is designed to restore plant equilibrium without reactor trip following a 10 percent step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15 percent and 100 percent full load, the power range for automatic reactor control. In effect, during load change conditions the reactor control system attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed setpoint at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step decrease in turbine load, the secondary side steam pressure and temperature initially increase. During this time, the RCS average temperature and pressurizer pressure also increase, but this change lags slightly behind the secondary side changes. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With the step load decrease, the reactor coolant temperature will ultimately be reduced from its peak value to a value below its initial equilibrium value. The reactor coolant average temperature setpoint change is made as a function of turbine generator load as determined by first stage turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash which reduces the rate of pressure decrease. Subsequently the pressurizer heaters come on to restore the pressurizer pressure to its normal value.

Following a step increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature will be raised to a value above its initial equilibrium value at the beginning of the transient.

The number of each operation is specified at 2000 times or 50 per year for the 40-year plant design life.

The 10 percent step load increase transient for the change from 90 percent to 100 percent of full load can be initiated at any power level between 15 percent and 90 percent full load. The 10 percent step load decrease transient for the change from 100 percent to 90 percent of full load can be initiated at any power level between 25 percent and 100 percent of full load.

G. Large Step Load Decrease With Steam Dump

This transient applies to a step decrease in turbine load from full power of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate a secondary side steam dump that will prevent both reactor trip and lifting of steam generator and pressurizer safety valves. The plant is designed to accept a step decrease of (later) percent of full power (complete loss of outside load, but retaining the plant auxiliary load) the steam dump system provides a heat sink to accommodate the difference in allowable unloading rates between the turbine and the RCS.

Subsequent to the large step load decrease, reactor power is reduced at a controlled rate, resulting in lower flow through the steam dump system. Another consequence of this event is turbine overspeed to 110 percent of nominal (controlled overspeed just below the turbine overspeed trip setpoint). This results in proportional increases in generator bus frequency, reactor coolant pump speed, and reactor coolant flowrate.

The number of occurrences of this transient is specified at 200 times or 5 per year for the 40-year plant design life.

H. Load Regulation

The plant is designed to participate actively in the minute-to-minute load sharing duty for the electrical network. The function of the load regulation mode of operation is to minimize the power production costs of the utility. Power allocations are calculated on a continuous basis (generally by a dispatch computer) and the outputs of the various units on the grid are adjusted automatically to minimize costs.

The normal load variations can be accommodated by operation of the Reactor Control System. However, this could lead to excessive control rod and/or mechanism wear. It also could generate axial power distributions which might require unusual operator attention or excessive adjustments in reactor coolant boron concentrations. These effects can be minimized by operation in the load regulation mode, under which normal reactor power level variations are controlled only by changes in coolant temperature. These are converted into reactor power changes through negative moderator coefficient feedback.

Load regulation, in response to changes in electrical network demand as described above, is referred to as automatic frequency control (AFC). In addition, load regulation may be employed in connection with governor free operation (GFO). GFO involves small but frequent changes in demand initiated at the turbine level with the turbine control system regulating the throttle valves in response to grid frequency deviations. It is likely that during most of the plant life the turbine will operate under both AFC and GFO inputs.

Load regulation is accomplished simply by expanding the temperature error deadband in the Rod Control System. this will allow the coolant temperature to drift (to supply reactivity changes through the moderator temperature coefficient) in response to the locally or remotely initiated power changes.

Whether or not the plant is to participate in load regulation and the allowable magnitude of the load variations due to AFC and GFO operation are set by the plant operator. For purposes of this design transient, it is assumed that the plant participates in the maximum allowable level of load regulation at all times during power operation with the only restriction being that the combination of nominal operating power level and maximum load regulation variations does not result in the plant exceeding 100% rated load. The maximum allowable level of load regulation varies during core life. A larger level of load regulation is permitted later in core life as the moderator temperature coefficient becomes more negative. For design purposes it is assumed that the allowable levels of AFC and GFO are 5 percent and 3 percent, respectively, of full power.

The allowable levels of AFC and GFO were established based on the following component cycling limits. Assuming continuous operation of the plant in the Load Regulation Mode for the 40-year plant life and accounting for 90% availability the following component cycling limits should not be exceeded:

(a) CRDM Stepping $\leq 10 \times 10^6$ steps
(b) Pressurizer Spray On-Off Cycling $\leq$ 500,000
(c) Pressurizer Backup Heater On-Off Cycling $\leq$ 500,000

The above noted component cycling should be considered to occur in addition to the duty cycles imposed on these components due to all other modes of plant operation.

I. Boron Concentration Equalization

Following significant increase in RCS boron concentration relative to pressurizer boron concentration, the pressurizer spray is initiated in order to equalize concentration between the loops and the pressurizer. This can be done by manually operating the pressurizer backup heaters, thus causing a pressure increase which will initiate spray. The pressurizer pressure increases above the setpoint value before being returned to 2250 psia by the proportional spray. This pressure is then maintained at 2250 psia by spray operation, matching the heat input from the backup heater until the concentration is equalized.

For design purposes, it is assumed that this operation is performed once during each daily design basis load follow cycle. With one load follow cycle per day and a 90 percent plant availability factor over the 40-year design life, the total number of occurrences is 13,200.

The only effects of these operations on the primary system are as follows:

o The reactor coolant pressure varies in step with the pressurizer pressure.

- The pressurizer surge line nozzle at the hot leg will experience the temperature shocks associated with outflow with the pressurizer.

These operations cause no significant effects on the steam generator secondary sides.

J. Feedwater Cycling

This transient can occur when the plant is at hot standby no-load conditions, during which intermittent feeding start-up feedwater into the steam generators is assumed. Due to fluctuations arising from this mode of operation, the reactor coolant average temperature decreases to a lower value and then immediately begins to return to normal no-load temperature. This transient is assumed to occur 2000 times over the life of the plant.

Feedwater additions required during plant heatup and cooldown operations are also assumed to be covered by the feedwater cycling transient, but with no increase in the total number of cycles. An occurrence (one cycle) is assumed to last two hours.

K. Loop Out of Service

For equipment design purposes, the plant is assumed to be operating at a reduced power level with a single loop out of service for limited periods of time. This is accomplished by reducing reactor power level and tripping a single reactor coolant pump (as opposed to tripping a pump while at full power, as discussed in the Partial Loss of Flow Transient description). Flow increase in the loops which remain in service (active loops) and reverse flow is established in the loop with the idle pump (inactive loop). Flow through the reactor is reduced.

When a loop is to be shut down, reactor power is reduced to the maximum allowable power level for N-1 loop operation and conditions stabilized. The pump is tripped and conditions are again stabilized at the same power level.

It is assumed that this transient occurs twice per year or 80 times in the life of the plant. Conservatively, it is assumed that all 80 occurrences can occur in the same loop. In other words, it must be assumed that the whole RCS is subjected to 80 transients while each loop is also subjected to 80 inactive loop transients.

When an inactive loop is brought back into service, the power level is reduced to approximately 10 percent, conditions stabilized, and the inactive reactor coolant pump is started. Subsequent return to full power is conducted in accordance with a normal loading operation. It is assumed that an inactive loop is inadvertently started while reactor power is above the allowable value 10 times over the life of the plant. (This transient is covered under upset conditions.) Thus, the normal start-up of an inactive loop is assumed to occur 70 times during the life of the plant.

L. Refueling

At the beginning of the refueling operation, the RCS is assumed to have been cooled to 140°F. At this time the vessel head is removed and the refueling canal is filled. This is done by transferring water from the emergency water storage tank into the loops by means of the residual heat removal (RHR) pumps. The refueling water flows directly into the reactor vessel by way of the accumulator connections and cold legs.

For design purposes this operation is assumed to occur twice per year or 80 times over the life of the plant.

M. Turbine Roll Test

This transient occurs during the hot functional test period for turbine cycle checkout. Reactor coolant pump power is used to heat the reactor coolant to operating temperature (no-load conditions) and the steam generated will be used to perform a turbine roll test. However, the plant cooldown during this test will exceed the 100°F per hour design rate.

The number of such test cycles is specified as 20, to be performed at the beginning of plant operating life prior to reactor operation. This transient occurs before plant start-up and the number of cycles is, therefore, independent of other operating transients.

N. Primary Side Leakage Test

A leakage test will be performed after each opening of the primary system. During this test the primary system pressure is raised (for design purposes) to 2500 psia, with the system temperature above the minimum temperature imposed by reactor vessel material ductility requirements, while the system is checked for leaks.

In actual practice, the primary system is pressurized to approximately 2400 psig, as measured at the pressurizer, to prevent the pressurizer safety valves from lifting during the leakage test. In addition, the secondary side of the steam generator must be pressurized so that the pressure differential across the tube sheet does not exceed 1600 psi. This can be accomplished with the steam, feedwater and blowdown lines closed off if the plant is cold. However, this test is usually conducted toward the end of a normal plant heatup, so the secondary side is hot and pressurized and this limitation can be met without difficulty.

For design purposes it is assumed that 200 cycles of this test will occur during the 40-year design life of the plant.

O. Secondary Side Leakage Test

During the life of the plant it may be necessary to check the secondary side of the steam generator, particularly the manway closures, for leakage. For design purposes, it is assumed that the steam generator secondary side is pressurized to just below its design pressure, to prevent the safety valves from lifting. In order not to exceed a secondary side to primary side pressure differential of 670 psi, the primary side must also be pressurized. The primary system must be above the minimum temperature imposed by reactor vessel material ductility requirements, typically between 120°F (BOL) and 250°F (EOL). It is assumed that this test is performed 80 times during the life of the plant.

P. Core Lifetime Extension

This transient can occur at the end of normal core life when the upper limit on boron concentration for maintaining full thermal power condition becomes less than can reasonably be achieved by dilution. In order to extend core life time the operator will:

1) Allow the reactor coolant average temperature to decrease below the normal programmed value, thereby compensating for fuel burnup through the negative moderator temperature coefficient of reactivity.

2) Manually control the turbine to maintain full thermal power conditions until the turbine throttle valves have fully opened.

3) Reduce turbine load by the amount necessary to maintain adequate reactor operating margin during the brief period of time necessary to take a feedwater heater out of service.

4) Take one feedwater heater out of service.

5) Increase turbine load to the full thermal power value.

This process can be repeated until several banks of feedwater heaters have been removed from service and the turbine throttle values are fully open. The transient is then completed.

For design purposes this transient is assumed to occur once per year for a total of 40 occurrences over the plant design life. During this mode of operation the plant is not capable of daily load follow. Thus, this transient must be considered separately from the Unit Loading and Unloading and Reduced Temperature Return to Power transients.

Q. Feedwater Heaters Out of Service

During the course of plant operating, one or more feedwater heaters may be taken out of service. During the period of time that the heaters are out of service, it is desirable to maintain the plant at full rated thermal load. To accomplish this:

1) Reduce turbine load by the amount necessary to maintain adequate reactor operating margin during the brief period of time necessary to take the feedwater heaters out of service.

2) Take the designated feedwater heaters out of service.

3) Increase turbine load to the full thermal power value.

The specific case considered is both high pressure feedwater heaters out of service. This is the most conservative case to evaluate as it produces larger temperature changes than other feedwater heater outage cases. This case is also the most credible, as it allows full power to be realized.

For design purposes, it is assumed that this transient occurs 3 times per year or 120 times over the life of the plant.

3.9.1.1.2 Level B Service Conditions (Upset Conditions)

The following primary system transients are considered upset conditions (Level B Service Conditions):

A. Loss of load

B Loss of offsite power

C. Partial loss of flow

D. Reactor trip from low power

Case A - With No Cooldown

Case B - With No Cooldown and No S.I.

Case C - With No Cooldown and S.I.

E. Reactor trip from full power

F. Inadvertent RCS depressurization

G. Inadvertent start-up of an inactive loop

H. Control rod drop

I. Excessive feedwater flow

J. Cold overpressurization

K. Sudden stoppage of flow

L. Operating basis earthquake

A. Loss of Load

This transient applies to a step decrease in turbine load from full power (turbine trip) without immediately initiating a reactor trip. This represents the most severe pressure transient on the RCS under Upset conditions. The reactor eventually trips as a consequence of a high pressurizer level trip initiated by the reactor protection system (RPS). Since redundant means of tripping the reactor are provided as a part of the RPS, a transient of this nature is not expected but is included to ensure a conservative design.

The number of occurrences of this transient is specified at 40 times or once per year for the 40-year plant design life.

B. Loss of Offsite Power

This transient applies to a blackout situation involving the loss of outside electrical power to the station, assumed to be operating initially at 100 percent power, followed by reactor and turbine trips. Under these circumstances, the reactor coolant pumps are deenergized, as are all electrical loads connected to the turbine-generator bus including the main feedwater and condensate pumps. Following coastdown of the reactor coolant pumps, natural circulation builds up in the system to some equilibrium valve. This condition permits removal of core residual heat through the steam generators which at this time are being cooled by the Secondary Side Safeguards System. Steam is removed for reactor cooldown through the steam generator safety valves.

The number of occurrences of this transient is specified at 40 times or once per year for the 40-year plant design life.

This transient also bounds the Emergency Condition Complete Loss of Flow transient, as well as loss of main feedwater. The postulated number of occurrences of the Loss of Offsite Power transient is considered to cover all three events.

C. Partial Loss of Flow

This transient applies to a partial loss of flow from full power, in which a reactor coolant pump is tripped out of service as the result of a loss of power to that pump. The consequences of such an event are a reactor on low reactor coolant flow, followed by turbine trip and automatic opening of the steam dump system. Flow reversal occurs in the affected loop which causes reactor coolant at cold leg temperature to pass through the steam generator and be cooled still further. This cooled water then flows through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizable reduction in the hot leg coolant temperature of the affected loop.

The number of occurrences of this transient is specified at 40 times or once per year for the 40-year plant design life.

D. Reactor Trip from Low Power

A significant fraction of the reactor trips experienced by an operating plant will occur during normal power loading and unloading operations, with the highest probability existing during a normal plant start-up with reactor power in the 10-25 percent range. Most probable causes of these trips are low steam generator water level, instrument error, and operator error(e.g., failure to execute prescribed manual blocks of nuclear instrumentation trips) during loadinq.

Since the initial primary-secondary temperature difference is less than at full power conditions, the RCS cooldown transient is much less severe than that accompanying a reactor trip from full power.

The definition of a low power reactor trip transient is consistent with operating plant experience and avoids the "fatigue analysis penalty" associated with bounding these events by trips occurring at full power.

For design purposes this transient is postulated to occur 200 times during the plant lifetime, and to be initiated when the reactor is operating at approximately 25 percent power.

E. Reactor Trip From Full Power

A reactor trip from full power may occur from a variety of causes resulting in temperature and pressure transients in the RCS and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant through the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of steam remove the core residual heat and prevent steam generator safety valve actuations. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the reactor protection system (RPS) causes the control rods to move into the core.

For design purposes, reactor trip is assumed to occur a total of 200 times or 5 times per year over the life of the plant. Three (3) basic cooldown cases are considered.

Case A - Reactor Trip with No Cooldown

Steam and feedwater flow are both controlled to bring the plant back to the no-load conditions and maintain it at no-load. It is assumed that the turbine control system operates as designed in 95 percent of the 200 reactor trip cases. For the remaining 5 percent, or 10 occurrences, is is conservatively assumed that this system fails, resulting in an emergency turbine overspeed. This situation could be initiated with malfunction of the turbine control system following a Large Step Load Decrease with Steam Dump, resulting in turbine speed increase past the overspeed trip set point. It is assumed that the reactor then trips and that the turbine speed increases to 120 percent of nominal, with accompanying proportional increases in generator bus frequency, reactor coolant pump speed, and reactor coolant flow rate. None of the other RCS primary side, pressurizer, or steam generator secondary side variables are affected.

For design purposes, it is assumed that the Emergency Turbine Overspeed constitutes a special case of the Reactor Trip with No Cooldown transient. Thus, for 10 of the 80 occurrences the effects of the reactor coolant flow variation are to be considered in addition to the basic pressure and temperature variations.

Case B - Reactor Trip with Cooldown and No Safety Injection

For this case, it is assumed that start-up feedwater system is actuated on low steam generator water level and that both start-up and main feedwater flow continues for approximately one minute after the reactor trip, maintaining a high heat transfer rate through the steam generator. This continues to drive the primary side pressure and temperature down and RCS pressure decreases to just above the safety injection setpoint. The main feedwater flow is terminated while startup feedwater flow is continued. The plant is brought back to no-load conditions. For design purposes, 80 occurrences of this transient are specified.

Case C - Reactor Trip with Cooldown and Safety Injection

This transient is similar to Case B, but it is assumed that the protection system setpoints are such that the RCS pressure decreases to just below the safety injection setpoint. The high head safety injection system is actuated; its operation lowers the RCS temperature and raises the RCS pressure. After approximately one minute, main feedwater flow is terminated while emergency feedwater flow (actuated on the safety injection signal) is continued. The plant is brought back to the no-load condition after safety injection is manually terminated. For design purposes, 40 occurrences of this transient are specified.

This transient is considered to bound the Upset Condition Inadvertent Safety Injection Actuation transient, as the consequences of a Case C reactor trip and an inadvertent S.I. are essentially the same. Inadvertent S.I. actuation will cause a reactor trip, which is assumed to be accompanied by excessive cooldown (Case C). The postulated number of occurrences covers both events.

F. Inadvertent Reactor Coolant System Depressurization

Several events can be postulated as occurring during normal plant operation which will cause rapid depressurization of the RCS. These include:

a. Actuation of a single pressurizer safety valve.

b. Inadvertent opening of one pressurizer power-operated relief valve due either to equipment malfunction or operator error.

c. Malfunction of a single pressurizer pressure controller causing one power-operated relief valve and two pressurizer spray valves to open.

d. Inadvertent opening of one pressurizer spray valve, due either to equipment malfunction or operator error.

e. Inadvertent auxiliary spray.

Of these events, the pressurizer safety valve actuation causes the most severe transients, and is used as a conservative "umbrella" case to represent the reactor coolant pressure and temperature variations arising from any of them.

a. Umbrella Case

When a pressurizer safety valve opens and remains open, the system rapidly depressurizes, the reactor trips, and the safety injection system (SIS) is actuated. The passive accumulators of the SIS are actuated when pressure decreases to approximately 1600 psi. The RCS reaches an equilibrium condition where the water release rate through the open pressurizer safety valve is equivalent to the safety injection flow. The RCS is also cooled down by the flow through the safety valve, the safety injection flow, and emergency feedwater flow.

Eventually, the plant must be taken to a cold shutdown condition, as the operator can take no immediate action to stop the transient and bring the plant to hot standby if the safety valve remains open.

For design purposes, this transient is assumed to occur 10 times during the 40-year design life of the plant.

b. Inadvertent Auxiliary Spray

Although inadvertent auxiliary spray actuations are included among the transient events covered by the umbrella case, the pressurizer safety valve actuation case selected to represent all the depressurization transients does not involve spray operation. Therefore, for the umbrella case it is assumed that pressurizer spray is not actuated, and that no temperature transients due to flow occur at the spray nozzle.

However, should auxiliary spray flow be initiated inadvertently, it could cause severe thermal shock at the pressurizer spray nozzle and on the pressurizer vessel. Therefore, to ensure a conservative design for these components, an "inadvertent auxiliary spray" transient is defined.

The inadvertent auxiliary spray transient will occur if the auxiliary spray valve is opened during normal plant operation due to failure of a control component or operator error. This will introduce cold water into the pressurizer resulting in a sharp pressure decrease and eventually in a low pressure reactor trip. The temperature of the auxiliary spray flow is dependent upon the performance of the regenerative heat exchanger. The most conservative case assumes that the let down stream is shut off, and that unheated charging fluid enters the 653°F pressurizer. For design purposes, it is assumed that the temperature of the spray water is 70°F and that the spray flowrate is equal to the normal charging rate. It is also assumed that auxiliary spray flow continues for five minutes before it is shut off, and that the resulting 653°F and 70°F temperature changes at the pressurizer and spray nozzle occur as steps.

The resulting rapid RCS pressure reduction is less severe than that accompanying the umbrella case transient and need not be considered separately. Also for design purposes, it is assumed that no reactor coolant temperature changes occur as the result of inadvertent auxiliary spray.

The total number of occurrences of this transient during the 40-year design life of the plant is specified as ten.

G. Inadvertent Start-up of an Inactive Loop

This transient can occur when a loop is out of service as described in Subsection 3.9.1.1.1, Loop Out of Service. With the plant operating at maximum allowable power level, the reactor coolant pump in the inactive loop is started as a result of operator error. Reactor trip occurs on high nuclear flux. For design purposes, this transient is assumed to occur ten times during the life of the plant.

H. Control Rod Drop

This transient occurs if a bank of control rods drops into the fully inserted position due to a single component failure. The reactor is tripped on either low pressurizer pressure or negative flux rate, depending on time in core life and magnitude of the reactivity insertion. It is assumed that this transient occurs 40 times over the life of the plant.

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6″

I. Excessive Feedwater Flow

This transient is conservatively defined as an umbrella case to cover occurrence of several events of the same general nature.

The postulated excessive feedwater flow transient results from inadvertent opening of a feedwater control valve when the plant is at the hot standby or no-load condition. The feedwater, condensate, and heater drains systems are in operation. The stem of a feedwater control valve is assumed to fail with the valve immediately reaching the full open position. In the affected steam generator (failed loop), the feedwater flow step increases from essentially zero flow to the value determined by the system resistance and the developed head of all operating feedwater pumps. Steam flow is assumed to remain at zero and main feedwater flow is isolated on a reactor coolant low T_{avg} signal; a subsequent lower T_{cold} signal (or a low pressurizer pressure signal) actuates the Integrated Safeguards. Feedwater flow is initiated by the safety injection signal, and it is assumed that all EFW pumps discharge 32°F water into the affected steam generator. It is assumed for conservatism in the secondary side analysis that emergency feedwater flows to the steam generators not affected by the malfunctioned valves in these "unfailed loops." After plant conditions stabilize, the emergency feedwater flow is terminated. The plant is then either taken to cold shutdown, or returned to the no-load condition at a nominal heatup rate using the start-up Feedwater System under manual control. For design purposes, this transient is assumed to occur 30 times during the 40-year life of the plant.

J. Cold Overpressurization

RCS cold overpressurization events are important from the standpoint of brittle fracture and crack propagation in an irradiated reactor vessel. In addition, other (non-irradiated) RCS components may be affected by application of high pressure at temperatures below the NDT temperature.

Of concern primarily are RCS pressurization rates and the effects of cold overpressurization mitigation controls on the maximum and minimum pressures reached following initiation of the transient.

One or more of the following basic types of cold overpressurization mechanisms should be considered in the analysis of RCS components.

- o Heat input mechanism, represented by start-up of a single reactor coolant pump immediately following a plant cooldown with water in the steam generators being warmer than the water in the reactor and loop piping. Pump operation will circulate the colder loop water through the steam generator, resulting in heating of this water and a rapid RCS pressure increase.

- o Mass input mechanism, represented by letdown line isolation while charging flow continues with a water solid pressurizer. The net mass addition results in rapid pressurization of the RCS.

- o Mass input mechanism, represented by inadvertent start-up of a single safety injection pump, with a water solid pressurizer and continuation of normal charging and letdown. The net mass addition results in rapid pressurization of the RCS. (The probability of occurrence of this type of event may be extremely low due to administrative controls and/or control system features.)

Each of these cases should be evaluated at both the low end (70°F) and the high end (300-350°F) of the brittle fracture temperature range.

For design purposes 10 occurrences are defined, of the applicable most conservative case for each RCS component sensitive to cold overpressurization effects.

K. Sudden Stoppage of Flow

This event is based on the instantaneous seizure of a reactor coolant pump rotating assembly while the plant is operating at full power. The resulting reactor coolant pressure, temperature, and flow variations will be the same as for the Condition IV Reactor Coolant Pump Locked Rotor transient. The analyses of the affected RCS components, however, must show that stress levels remain within Level B Service limits for this Upset Condition event. An exception is the affected reactor coolant pump, for which the analyses of pressure boundary components must satisfy Level D Service limits. (It is assumed that the affected pump would be replaced.)

For design purposes this transient is assumed to occur five times during the plant lifetime.

L. Operating Basis Earthquake (OBE)

The OBE is that earthquake which is reasonably expected to occur during the plant life. The number of occurrences for fatique evaluation is assumed to be five earthquakes each with $\leq$ 0.1 g horizontal ZPA with Regulatory Guide 1.60 spectra.

3.9.1.1.3 Level C Service Conditions (Emergency Conditions)

The following reactor coolant system transients are considered emergency conditions (Level C Service Conditions):

A. Small loss-of-coolant accident
B. Small steam line break
C. Small feedwater line break.

A. Small Loss-of-Coolant Accident

For design transient purposes, the small loss of coolant accident is defined as a break equivalent to the severance of a 1 inch inside diameter branch connection. Breaks smaller than one inch ID are also covered by this definition. (The one inch and smaller breaks do not cause accumulator injection; those which are much larger than one inch will cause accumulator injection and are considered Faulted Conditions. Breaks smaller than approximately 3/8 inch produce no significant thermal transients and can be handled by the normal makeup system.) The Reactor Coolant system depressurizes quickly, and it is assumed that the Integrated Safeguards System is actuated as its pressure setpoint is reached. It delivers water at an assumed minimum temperature of 32°F as soon as RCS pressure falls below the shutoff heat of the high head SI pumps.

For design purposes, it is assumed that this transient occurs five times during the life of the plant.

B. Small Steam Line Break

For design transient purposes, a small steam line break is defined as a break equivalent in effect to a steam generator safety valve opening and remaining open. The following conservative assumptions are used in defining the transients:

a. The reactor is initially in a hot, zero power condition.

b. The small steam line break results in immediate reactor trip and SI actuation.

c. A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient.

Operation of the high head safety injection pumps repressurizes the RCS within a relatively short time to the pressure corresponding to the shutoff head of the pumps.

This transient is assumed to occur five times during the life of the plant.

C. Small Feedwater Line Break

This transient is postulated to result from rupture of an emergency feedwater line or other small line, or from a small break in the feedwater line itself, between the feedwater isolation valve and the steam generator. The reactor is tripped, either by a safety injection signal generated on low steam line pressure or on low steam generator water level. The Emergency Feedwater System is actuated on low steam generator water level and delivers cold emergency feedwater to the intact steam generators. The affected steam generator eventually blows down following the break.

Cases both with and without reactor coolant pump operation are considered. For design purposes five occurrences of this transient during the plant lifetime are postulated.

3.9.1.1.4 Level D Service Conditions (Faulted Conditions)

The following reactor coolant system transients are considered faulted conditions (Level D Service Conditions). Each of the following accidents should be evaluated for one occurrence:

A. Primary coolant system auxiliary line pipe break (large loss-of-coolant accident)
B. Large steam line break
C. Feedwater line break
D. Reactor coolant pump locked rotor
E. Control rod ejection
F. Steam generator tube rupture
G. Safe shutdown earthquake

A. Primary Coolant System Auxiliary Line Pipe Break (Large Loss-of-Coolant Accident)

Following rupture of a primary coolant system auxiliary line pipe resulting in a large loss of coolant, the primary system pressure decreases causing the primary system temperature to decrease. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal will still be at or near the operating temperature by the end of blowdown. The Integrated Safeguards System is actuated on low pressurizer pressure to introduce water from the EWST (at an assumed minimum temperature of 80°F) into the RCS. The safety injection signal also results in reactor and turbine trips.

B. Large Steam Line Break

This transient is based on the complete severance of the largest steam line. The following conservative assumptions were made:

a. The reactor is initially in a hot, zero power condition.

b. The steam line break results in immediate reactor trip and SI actuation of the Integrated Safeguards System. The emergency feedwater system is actuated.

  o No return to criticality occurs in the core.

  o The Integrated Safeguards and Emergency Feedwater Systems operate at design capacity to increase the cooldown rate (both systems) and repressurize the Reactor Coolant System (Integrated Safeguards System).

  As a loss of power could occur at the same time as the steam line break, two cases should be considered:

  o With offsite power -- "Pumps running" case -- Reactor coolant pumps remain in operation.

- o Without offsite power -- "pumps tripped" case -- Reactor coolant pumps are de-energized and coolant flow coasts down to the natural circulation value.

The designer must evaluate both of these cases and select the one which represents the most conservative design values for the particular component.

C. Feedwater Line Break

This accident involves a double-ended rupture of the main feedwater piping from full power, resulting in the rapid blowdown of one steam generator and the termination of main feedwater flow to the others. Turbine trip, with immediate reactor trip, occurs on a low-low level signal from the faulted steam generator. The emergency feedwater system (EFWS) is actuated and cools a minimum of two intact steam generators. Loss of the plant from the grid is assumed to cause a blackout, all RCPs are deenergized and coast down to reduce the coolant flow to the natural circulation value. The Integrated Safeguards System is actuated and is assumed to deliver maximum safeguards flow until manually shut off.

In the analysis no credit is taken for operation of pressure control systems, steam dump or steam generator power operated relief valves, and it is assumed that steam line check valves are not provided. Thus, the intact steam generators feed the break through the main steam header after the steam generator with the break discharges its liquid inventory. Steam flow continues until the main steam lines are isolated on low steam line pressure. The magnitudes of the reverse steam and liquid flows from the intact steam generators to the break are dependent on the minimum flow area in the feedwater ring and feedwater nozzle.

D. Reactor Coolant Pump Locked Rotor

This accident is based on the instantaneous seizure of a reactor coolant pump with the plant operating at full power. The locked rotor can occur in any loop. Reactor trip occurs almost immediately as the result of low coolant flow in the affected loop.

E. Control Rod Ejection

This accident is based on the single most reactive control rod being instantaneously ejected from the core. This reactivity insertion in a particular region of the core causes a severe pressure increase in the RCS such that the pressurizer safety valves will lift and also causes a more severe temperature transient in the loop associated with the affected region (the so-called "hot" loop) than in the other loops. For conservatism the analysis is based on the reactivity insertion and does not include the mitigating effects (on the pressure transient) of coolant blowdown through the hole in the vessel head vacated by the ejected rod.

F. Steam Generator Tube Rupture

This accident is postulated as the double-ended rupture of a single steam generator tube resulting in decreases in pressurizer level and reactor coolant pressure. Eventually the loss of reactor coolant causes a reactor trip (also a turbine trip) on low pressurizer pressure. The ensuing plant cooldown results in Integrated Safeguards System actuation due to low pressurizer pressure. The safety injection signal automatically starts the emergency feedwater pumps and isolates the main feedwater lines. The steam line leading from the affected steam generator is isolated. When the pressurizer water level is recovered, the operator stops safety injection and cold shutdown conditions.

For the RCS primary side, this event will cause a transient which is no more severe than that associated with Reactor Trip with Cooldown and Safety Injection. Therefore, no special primary side stress analysis is required. An analysis of the secondary side will be prepared as part of the FDA when detailed requirements have been developed for the steam generator and related RCS and plant designs.

G. Safe Shutdown Earthquake

The SSE is defined as the maximum vibratory ground motion which can reasonably be predicted from geologic and seismic evidence. For design purposes this is assumed as $\leq$ 0.3 g horizontal ZPA with Regulatory Guide 1.60 Spectra.

3.9.1.1.5 Test Conditions

The following reactor coolant system transients under test conditions are discussed:

A. Primary side hydrostatic test
B. Secondary side hydrostatic test
C. Tube leakage test

A. Primary Side Hydrostatic Test[1]

The pressure tests include both shop and field hydrostatic tests which occur as a result of component or system testing. This hydro test is performed at a water temperature which is compatible with reactor vessel material ductility requirements and a test pressure of 3107 psig (1.25 times design pressure). In this test, the RCS is pressurized to 3107 psig coincident with steam generator secondary side pressure of 0 psig. The RCS is designed for 10 cycles of these hydrostatic tests which are performed prior to plant start-up. The number of cycles is independent of other operating transients.

Additional hydrostatic tests will be performed to meet the in-service inspection requirements of ASME Section XI. A total of four such tests is expected. The increase in the fatigue usage factor caused by these tests

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[1]These hydrostatic test cycles are to be considered in the stress and fatigue analyses.

is easily covered by the conservative number (200) of primary side leakage tests that are considered for design and no additional specification is required.

B. Secondary Side Hydrostatic Test[1]

The secondary side of the steam generator is pressurized to 1.25 design pressure with a minimum water temperature of 120°F coincident with the primary side at 0 psig.

For design purposes, it is assumed that the steam generator will experience 10 cycles of this test.

These tests may be performed either prior to plant start-up, or subsequently following shutdown for major repairs or both. The number of cycles is therefore independent of other operating transients.

C. Tube Leakage Test

During the life of the plant, it may be necessary to check the steam generator for tube leakage and tube-to-tubesheet leakage. This is done by visual inspection of the underside (channel head side) of the tube sheet for water leakage with the secondary side pressurized. Tube leakage tests are performed during plant cold shutdowns.

For these tests the secondary side of the steam generator is pressurized with water, initially at a relatively low pressure, and the primary system remains depressurized. The underside of the tube sheet is examined visually for leaks. If any are observed, the secondary side is then depressurized and repairs made. The secondary side is then repressurized (to a higher pressure) and the underside of the tube sheet is again checked for leaks. This process is repeated until all the leaks are repaired. The maximum (final) secondary side test pressure reached is 840 psig.

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[1]These hydrostatic test cycles are to be considered in the stress and fatigue analyses.

Both the primary and secondary sides of the steam generator are be at ambient temperature during these tests.

The total number of tube leakage test cycles is defined as 800 during the 40-year life of the plant. Following is a breakdown of the anticipated number of occurrences at each secondary side test pressure:

Test Pressure, psig	Number of Occurrences
200	400
400	200
600	120
840	80

Neither the primary side nor secondary side design pressures are exceeded during the Tube Leakage Test. This test is included under Test Conditions since the expected secondary-to-primary pressure differential exceeds the design value of 670 psi for some of the test cycles.

3.9.1.2 Computer Programs Used in Analysis

3.9.1.2.1 NPB Systems and Components

The following computer programs will be used in dynamic and static analyses to determine mechanical loads, stresses, and deformations of Seismic Category I components and equipment.

A. WESTDYN - static and dynamic analysis of redundant piping systems.

B. FATCON - fatigue analysis of piping systems

C. WESAN - reactor coolant loop equipment support structures analysis and evaluation.

D. WECAN - finite-element structural analysis and nonlinear time history seismic analysis.

3.9.1.3 Experimental Stress Analysis

No experimental stress analysis methods are used for Category I systems or components. However, Westinghouse makes extensive use of measured results from prototype plants and various scale model tests as discussed in Subsection 3.9.2.

3.9.1.4 Consideration for the Evaluation of the Faulted Condition

The analytical methods used to evaluate the faulted conditions for seismic Category I ASME Code and non-code items are described in the Subsection 3.9.4 of this module.

3.9.2 Dynamic Testing and Analysis

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

A preoperational test program is implemented, as required by NB-3622.3, NC-3622, and ND-3611 of Section III of the ASME Boiler and Pressure Vessel Code, to verify that the piping and piping restraints will withstand dynamic effects due to transients such as pump trips and valve trips, and that piping vibrations are within acceptable levels.

The preoperational test program for the Class 1, 2, and 3 and high-energy piping systems is to simulate actual operating modes to demonstrate that the components comprising these systems meet functional design requirements and that piping vibrations are within acceptable levels. Piping systems are checked in three sequential steps or series of tests and inspections.

Construction acceptance, the first step, entails inspection of components for correct installation. During this phase, pipe and equipment supports are checked for correct assembly and setting. The cold locations of reactor coolant system (RCS) components, such as steam generators and reactor coolant pumps, are recorded.

During the second step of testing, plant heatup, the plant is heated to normal operating temperatures. During the heatup, all systems are observed periodically to verify proper expansion; expansion data is recorded at the end of heatup.

During the third step of testing, performance testing, systems are operated and performance of critical pumps, valves, controls, and auxiliary equipment is checked. This phase of testing includes transient tests such as reactor coolant pump trips, reactor trip, and relief valve testing. During this phase of testing, the piping and piping restraints are observed for vibration and expansion response. Automatic safety devices, control devices, and other major equipment are observed for indications of overstress, excess vibration, overheating, and noise. Each system test includes critical valve operation during transient system modes.

The locations in the piping system selected for observation during the testing, and the respective acceptance criteria, are provided in the detailed preoperational vibration, thermal expansion, and dynamic effects test program plan. These are submitted to the Nuclear Regulatory Commission (NRC) at least 60 days prior to the initiation of the test program.

Provisions are made to verify the operability of essential snubbers by recording hot and cold positions. If vibration during testing exceeds the acceptance criteria, corrective measures are taken and the test rerun to demonstrate adequacy.

Should additional restraints be installed, piping rerouted, or other corrective action taken as a result of the preoperational piping test, the NRC is provided with documentation of such action. The analysis verifying that system response is within acceptable limits will be on file.

Vibratory dynamic loadings can be placed in two categories: (1) transient induced vibrations and (2) steady-state vibrations. The first is a dynamic system response to a transient, time dependent forcing function, such as fast valve closure, while the second is a constant vibration, usually flow induced.

A. Transient Response

Dynamic events falling in this category are anticipated operational occurrences. The systems and the transients to be included in the preoperational test program to verify the piping system are:

1. Main steam turbine stop valve trip
2. Main steam atmospheric dump valves opening
3. Main steam condenser dump valves opening
4. Steam generator power-operated relief valve opening
5. Main steam isolation valve closure
6. Main feedwater line check valve closure
7. Pressurizer power operated relief valve opening
8. Pressurizer vent opening
9. Reactor vessel head vent opening

For these types of transients, a time-dependent dynamic analysis is performed on the system. The stresses thus obtained are combined with system stresses resulting from other operating conditions in accordance with the criteria provided in Subsections 3.9.1 and 3.9.3.

Details of the program and the pipe monitoring scratch plates and strain gage locations, including the criteria for evaluation of data gained, are provided in the test procedures.

B. Steady-State Vibration

System vibration resulting from flow disturbances falls into this category. Positive displacement pumps may cause such flow variation and vibrations and, as such, will be reviewed. Such systems will be checked, including the charging systems.

Since the exact nature of the flow disturbance is not known prior to pump operation, no analysis is performed. A steady-state vibrational testing will be conducted by visual and local measurements.

The acceptance criteria is that the maximum measured amplitude shall not induce a stress in the piping system greater than one-half the endurance limit, as defined in Section III of the ASME B&PV Code.

When required, additional restraints are provided to reduce the stresses to below the acceptance criteria levels.

During the thermal expansion test, pipe deflections will be measured or observed at various locations based on the location of snubbers, hangers, and expected large displacements. One complete thermal cycle, i.e., cold position to hot position to cold position, will be monitored. Acceptance criteria for the thermal expansion test will be based on the movements established by thermal piping analysis and will verify that the piping system is free to expand thermally (i.e., piping does not bind or lock at spring hangers and snubbers nor interferes with structure or other piping).

The systems to be monitored are selected portions of:

- o Reactor cooling system
- o Main steam system
- o Main feedwater system
- o Chemical and volume control system
- o Residual heat removal (RHR) system
- o Containment spray system
- o Emergency core cooling system (ECCS)
- o Secondary side safeguards system
- o Steam generator blowdown system
- o Component cooling water system

3.9.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

Westinghouse utilizes analysis, test, or a combination of test and analysis for seismic qualification of equipment. Testing is the preferred method; however, analysis is utilized when one of the following conditions is satisfied:

A. The equipment is too large or the external loads, connecting elements, or appurtenances cannot be simulated with a shaker table test.

B. The only requirement that must be satisfied relative to the safety of the plant is the maintenance of structural integrity (mechanical equipment only).

C. The component represents a simple linear system or nonlinearities can be conservatively accounted for in the analysis.

The operability of Seismic Category I mechanical equipment must be demonstrated if the equipment is active; i.e., mechanical operation is relied on to perform a safety function. The operability of active Safety Class 2 and 3 pumps, active Safety Class 1, 2, or 3 valves and their respective drives, operators and vital auxiliary equipment is shown by satisfying the criteria given in Subsection 3.9.3.2.

Inactive Seismic Category I equipment such as heat exchangers, racks, and consoles are shown to have structural integrity during a seismic event by analysis satisfying the stress criteria applicable to the particular piece of equipment.

A list of Seismic Category I equipment is provided in Table 3.2-1.

The criteria used to decide whether dynamic testing or analysis should be used to qualify Seismic Category I mechanical equipment are as follows:

A. Analysis Without Testing

1. Structural analysis without testing is used if structural integrity alone can assure the design-intended function. Equipment which falls into this category includes:

   a. Ductwork
   b. Tanks and vessels
   c. Heat exchangers
   d. Filters
   e. Inactive valves

2. Rotational analysis without testing is used to qualify rotating machinery items where it must be verified that deformations due to seismic loadings will not cause binding of the rotating element to the extent that the component cannot perform its design-intended function.

   The seismic qualification of pumps is discussed more fully in Subsection 3.9.3.2.1. The procedure discussed therein applies, with some variations, to other items in this category.

3. Dynamic analysis without testing is used to qualify heavy machinery too large to be tested. It is verified that deformations due to seismic loadings will not cause binding of the moving parts to the extent that the component cannot perform its required safety function. Components which fall into this category include:

   o Pumps
   o Turbines
   o Generators
   o Fans
   o Diesr engines

B. Dynamic Testing

Dynamic testing is used for components with mechanisms that must change position in order to perform their required safety function. Such components include:

- o Electric motor valve operators
- o Valve limit switches
- o Similar appurtenances for other active mechanical equipment

The seismic qualification of Seismic Category 1 electrical equipment is discussed in Section 3.10.

C. Combinations of Analysis with Testing

A combination of analysis, static testing, and dynamic testing is used for seismic qualification of complex equipment. Such equipment includes:

1. Standby diesel-generators
2. Turbine-driven emergency feedwater pumps
3. Main steam and main feedwater isolation valves
4. Other active valves

The seismic qualification of active valves is discussed more fully in Subsection 3.9.3.2.

The acceptance criteria which are used are as follows:

1. Tests, when used, demonstrate that the component is not prevented from performing its required safety function during and after the test.

2. Analysis, when used for qualification of vessels, pumps, or valves, verifies that stresses do not exceed the allowables specified in Tables 3.9-3 and 3.9-5 and that deformations do not exceed those which

will permit the component to perform its design-intended function. The results of tests and analyses of safety-related mechanical equipment are available for inspection.

3.9.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

For a discussion of the dynamic response analysis of the reactor internals, see Subsection 3.9.2.3 of RESAR-SP/90 PDA Module 5, "Reactor System."

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

For a discussion of the preoperational flow-induced vibration testing of the reactor internals, see Subsection 3.9.2.4 of RESAR-SP/90 PDA Module 5, "Reactor System."

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

For a discussion of the dynamic system analysis of the reactor internals, see Subsection 3.9.2.5 of RESAR-SP/90 PDA Module 5, "Reactor System."

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

American Society of Mechanical Engineers (ASME) Class 1, 2, and 3 components and component supports are designed to an appropriate combination of plant conditions and design loadings. The plant conditions are design, normal, upset, emergency, and faulted conditions. The design loadings are pressure, temperature and deadweight loads.

The ASME Code Class components are constructed in accordance with the ASME B&PV Code, Section III requirements. For Code Class 1 components, very stringent requirements are imposed and are met. For Code Class 2 and 3 components, the requirements are less stringent but adequate, in accordance with the lower classification.

3.9.3.1.1 ASME Code Class 1 Components and Supports

Loading combinations for ASME Class 1 components and component supports are presented in Table 3.9-2 and stress limits for these components are given in Table 3.9-3. A detailed discussion of design transients for the NPB components is provided in Subsection 3.9.1.

The structural stress analyses performed on the ASME Class 1 components and supports consider the loadings specified as shown in Table 3.9-2. These loads result from thermal expansion, pressure, weight, operating basis earthquake (OBE), safe shutdown earthquake (SSE), design basis loss-of-coolant accident, and plant operational thermal and pressure transients.

3.9.3.1.1.1 Analysis of the Reactor Coolant Loop Piping and Supports

The loads used in the analysis of the reactor coolant loop piping are described in detail below.

A. Pressure

Pressure loading is identified as either membrane design pressure or general operating pressure, depending upon its application. The membrane design pressure is used in connection with the longitudinal pressure stress and minimum wall thickness calculations in accordance with the ASME Code as stated in Table 3.2-1.

The term operating pressure is used in connection with determination of the system deflections and support forces. The steady-state operating hydraulic forces based on the system initial pressure are applied as general operating pressure loads to the reactor coolant loop model at change in direction or flow area.

B. Weight

A deadweight analysis is performed to meet Code requirements by applying a 1.0 g load downward on the complete piping system. The piping is assigned a distributed mass or weight as a function of its properties. This method provides a distributed loading to the piping system as a function of the weight of the pipe and contained fluid during normal operating conditions.

C. Seismic

The input for the reactor coolant loop piping seismic analysis is in the form of three statistically independent orthogonal time-history accelerations. The earthquake accelerations for the horizontal directions are applied to the containment basemat simultaneously with the vertical acceleration in the vertical direction.

For the OBE and SSE seismic analyses, 5 and 8 percent critical damping, respectively, are used in the reactor coolant loop/supports system analysis. (See Figure 3.7-8)

Optional seismic analysis methods which may be used for low seismic plants include the uniform response spectra method and the multiple response spectra method.

D. Loss-of-Coolant Accident

Blowdown loads are developed in the reactor coolant loops as a result of transient flow and pressure fluctuations following a postulated pipe break in one of the reactor coolant loop auxiliary connections. Structural consideration of dynamic effects of postulated pipe breaks requires postulation of a finite number of break locations. Postulated pipe break locations are given in Section 3.6.

Time-history dynamic analysis is performed for these postulated break cases. Hydraulic models are used to generate time dependent hydraulic forcing functions used in the analysis of the reactor coolant loop for each break case.

E. Transients

The ASME Code, Section III requires satisfaction of certain requirements relative to operating transient conditions. Operating transients are summarized in Subsection 3.9.1.1.

To provide the necessary high degree of integrity for the RCS, the transient conditions selected for fatigue evaluation are based on conservative estimates of the magnitude and anticipated frequency of occurrence of the temperature and pressure transients resulting from various plant operation conditions.

The analytical methods used in obtaining the solution consist of the transfer matrix method and stiffness matrix formulation for the static structural analysis, the time-history integration, or response spectra methods, for seismic dynamic analysis, and time history integration analysis methods for effects of high-energy line pipe breaks.

The integrated reactor coolant loop/supports system model is the basic system model used to compute loadings on components, component supports, and piping. The system model includes the stiffness and mass characteristics of the reactor coolant loop piping and components, the stiffness of supports, the stiffnesses of auxiliary line piping which affect the system. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

A. Static

The reactor coolant loop/supports system model, constructed for the WESTDYN computer program, is represented by an ordered set of data which numerically describes the physical system. Figure 3.9-1 shows an isometric line schematic of this mathematical model. The steam generator and reactor coolant pump vertical and lateral support members are described in Section 5.4.14 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System."

The spatial geometric description of the reactor coolant loop model is based upon the reactor coolant loop piping layout and equipment drawings. The node point coordinates and incremental lengths of the members are determined from these drawings. Geometrical properties of the piping and elbows along with the modulus of elasticity E, the coefficient of thermal expansion α, the average temperature change from ambient temperature ΔT, and the weight per unit length are specified for each element. The primary equipment supports are represented by stiffness matrices which define restraint characteristics of the supports. The vertical thermal growth of the reactor vessel nozzle centerline and equipment support points are considered in the construction of the model for thermal analysis.

The model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained. These loads are incorporated int the overall load vector.

After all the sections have been defined in this manner, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix, the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system.

The static solutions for deadweight, thermal, and general pressure loading conditions are obtained by using the WESTDYN computer program.

B. Seismic

The model used in the static analysis is modified for the dynamic analysis by including the mass characteristics of the piping and equipment. In the time-history seismic analysis, the containment internals structure, and all of the piping loops are included in the system coupled model. The effect of the equipment motion on the reactor coolant loop/supports system is obtained by modeling the mass and the stiffness characteristics of the equipment in the overall system model.

The steam generator is typically represented by four discrete masses. The lowest mass is located at the intersection of the centerlines of the inlet and outlet nozzles of the steam generator. The second mass is located at the steam generator upper support elevation. The third mass is located at the top of the transition cone and the fourth at the steam outlet nozzle.

The reactor coolant pump is typically represented by a two discrete mass model. The lower mass is located at the intersection of the centerlines of the pump suction and discharge nozzles. The upper mass is located near the center of gravity of the motor.

The reactor vessel and core internals are typically represented by approximately ten discrete masses. The masses are lumped at various locations along the length of the vessel and along the length of the representation of the core internals.

The component upper and lower lateral supports are inactive during plant heatup, cooldown and normal plant operating conditions. However, these restraints become active due to the rapid motions of the reactor coolant loop components that occur from the dynamic loadings, and are represented by stiffness matrices and/or individual tension or compression spring members in the dynamic model. The analyses are performed at the full power condition.

The total response is obtained by modal superposition time history integration of the equations of motion. The results of the analysis are time-history forces and displacements. The time-history displacement response is then used in computing support loads and in performing the reactor coolant loop piping stress evaluation.

The details of the response spectra seismic analysis, which is also sometimes used, are described in Subsection 3.7.

C. Loss-of-Coolant Accident

The mathematical model used in the static analyses is modified for the loss-of-coolant accident analyses to represent the severance of the reactor coolant loop auxiliary piping at the postulated break locations. Modifications include addition of the mass at the break. The natural frequencies and eigenvectors are determined from this loop model.

The time-history bydraulic forces at the node points are combined to obtain the forces and moments acting at the corresponding structural lumped-mass node points.

The dynamic structural solution for the full power loss-of-coolant accident is obtained by using a Runge-Kutta integration technique and normal mode theory.

When elements of the system can only be represented as single acting members (tension or compression members), they are considered as nonlinear elements, which are represented mathematically by the combination of a gap, a spring, and a viscous damper. The force in this nonlinear element is treated as an externally applied force in the overall normal mode solution. Multiple nonlinear elements can be applied at the same node, if necessary.

The loss-of-coolant accident displacements of the reactor vessel are applied in time-history form as input to the dynamic analysis of the reactor coolant loop. The loss-of-coolant accident analysis of the reactor vessel includes all the forces acting on the vessel including internals reactions and loop mechanical loads. The reactor vessel analysis is described in Subsection 3.9.5.

The asymmetric external pressure loads on the RCP and steam generator resulting from the postulated pipe rupture and pressure buildup in the loop compartments are applied to the same integrated reactor coolant loop/supports system model used to compute loadings on the components, component supports, and reactor coolant piping, as discussed above. Jet impingement loads on the RCL piping, components, and supports resulting from postulated auxiliary line pipe ruptures are also applied to the RCL/support model. The response of the entire system is obtained for the various external loading cases from which the internal member forces and piping stresses are calculated.

The time-history displacement response of the loop is used in computing support loads and in performing stress evaluation of the reactor coolant loop piping.

D. Fatigue

Operating transients in a nuclear power plant cause thermal and/or pressure fluctuations in the reactor coolant fluid. The thermal transients cause time-varying temperature distributions across the pipe wall. These temperature distributions resulting in pipe wall stresses may be further subdivided in accordance with the Code into three parts, a uniform, a linear, and a nonlinear portion. The uniform portion results in general expansion loads. The linear portion causes a bending moment across the wall and the nonlinear portion causes a skin stress.

The transients as summarized in Subsection 3.9.1.1 are used to define the fluctuations in plant parameters. A one-dimensional finite difference heat transfer program is used to solve the thermal transient problem. The pipe is represented by at least fifty elements through the thickness of the pipe. The convective heat-transfer coefficient employed in this program represents the time varying heat transfer due to free and forced convection. The outer surface is assumed to be adiabatic while the inner surface boundary experiences the temperature of the coolant fluid. Fluctuations in the temperature of the coolant fluid produce a temperature distribution through the pipe wall thickness which varies with time. The average through-wall temperature, T_A, is calculated by integrating the temperature distribution across the wall. This integration is performed for all time steps so that T_A is determined as a function of time.

A load-set is defined as a set of pressure loads, moment loads, and through-wall thermal effects at a given location and time in each transient. The through-wall thermal effects are functions of time and can be subdivided into four parts:

1. Average temperature (T_A) is the average temperature through-wall of the pipe which contributed to general expansion loads.

2. Radial linear thermal gradient which contributes to the through-wall bending moment (ΔT_1).

3. Radial nonlinear thermal gradient (ΔT_2) which contributes to a peak stress associated with shearing of the surface.

4. Discontinuity temperature ($T_A - T_B$) represents the difference in average temperature at the cross sections on each side of a discontinuity.

Each transient is described by at least two load-sets representing the maximum and minimum stress state during each transient. The construction of the load-sets is accomplished by combining the following to yield the maximum (minimum) stress state during each transient.

a. ΔT_1

b. ΔT_2

c. $\alpha_A T_A - \alpha_B T_B$

d. Moment loads due to T_A

e. Pressure loads

This procedure produces at least twice as many load-sets as transients for each point.

For all possible load-set combinations, the primary-plus-secondary and peak stress intensities, fatigue reduction factors (K_e) and cumulative usage factors, U, are calculated. The FATCON-7 program is used to perform this analysis in accordance with the ASME Code, Section III, Subsection NB-3650.

The combination of load-sets yielding the highest alternating stress intensity range is first used to calculate the incremental usage factor. The next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having allowable cycles $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

3.9.3.1.1.2 Class 1 Auxiliary Branch Lines

The allowable stresses for ASME Code Class 1 components and supports are given in Table 3.9-3. All Class 1 components and supports are designed and analyzed for Levels A, B, and C Service Conditions, and corresponding service level requirements to the rules and requirements of the ASME Code, Section III. The analysis or test methods, and associated stress or load allowable limits that are used in evaluation of Level D Service Conditions are those that are defined in Appendix F of the ASME Code, Section III with the following supplementary option.

The analytical methods used to obtain the solution consist of the transfer matrix method and stiffness matrix formulation for the static structural analysis, and the response spectrum method for seismic dynamic analysis.

The integrated Class 1 piping and supports system model is the basic system model used to compute loadings on components, component and piping supports, and piping. The system models include the stiffness and mass characteristics of the Class 1 piping components, the reactor coolant loop, and the stiffness of supports that affect the system response. The deflection solution of the entire system is obtained for the various loading cases from which the internal member forces and piping stresses are calculated.

A. Static

The Class 1 piping system models are constructed for the WESTDYN computer program, which numerically describes the physical system. A network model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the characteristic stiffness matrix for the section. Using the transfer relationship for a section, the loads required to suppress all deflections at the ends of the section arising from the thermal and boundary forces for the section are obtained.

After all the sections have been defined in this manner, the overall stiffness matrix and associated load vector to suppress the deflection of all the network points is determined. By inverting the stiffness matrix the flexibility matrix is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system.

The support loads are also computed by multiplying the stiffness matrix by the displacement vector at the support point.

B. Seismic

The models used in the static analyses are modified for use in the dynamic analyses by including the mass characteristics of the piping and equipment.

The lumping of the distributed mass of the piping systems is accomplished by locating the total mass at points in the system that will appropriately represent the response of the distributed system. Effects of the primary equipment motion, that is, reactor vessel, steam generator, reactor coolant pump, and pressurizer, on the Class 1 piping system are obtained by modeling the mass and the stiffness characteristics of the primary equipment and loop piping in the overall system model. Alternately, the effects of the primary equipment and loop motion are represented by resonse spectra and anchor motions calculated at the connection points to the RCS loop.

The supports are represented by stiffness matrices in the system model for the dynamic analysis. Shock suppressors that resist rapid motions are also included in the analysis. The solution for the seismic disturbance employs the response spectra method. This method employs the lumped mass technique, linear elastic properties, and the principle of modal superposition.

The total response obtained from the seismic analysis consists of two parts: the inertia response of the piping system, and the response from differential anchor motions. The stresses resulting from the anchor motions are considered to be secondary and, therefore, are included in the fatigue evaluation.

C. Loss-of-Coolant Accident

The mathematical models used in the seismic analyses of the Class 1 lines are also used for RCL auxiliary pipe break effect analysis. To obtain the dynamic solution for auxiliary lines 6 inches and larger, and certain small-bore lines required for ECCS consideration, the time history deflections from the analysis of the reactor coolant loop are applied at branch nozzle connections. For other small-bore lines that must maintain structural integrity, the motion of the RCL is applied statically.

D. Fatigue

A thermal transient heat transfer analysis is performed for each different piping component on all the Class 1 branch lines. The normal, upset, and test condition transients identified in Section 3.9.1.1 are considered in the fatigue evaluation.

For each thermal transient, two load sets are defined representing the maximum and minimum stress states for that transient.

The FATCON computer program is used to calculate the primary-plus-secondary and peak stress intensity ranges, fatigue reduction factors, and cumulative usage factors for all possible load set combinations. It is conservatively assumed that the transients can occur in any sequence, thus resulting in the most conservative and restrictive combinations of transients.

The combination of load sets yielding the highest alternating stress intensity range is determined and the incremental usage factor calculated. Likewise, the next most severe combination is then determined and the incremental usage factor calculated. This procedure is repeated until all combinations having an allowable cycle of $<10^6$ are formed. The total cumulative usage factor at a point is the summation of the incremental usage factors.

3.9.3.1.1.3 Loading Combinations and Stress Limits

Loading combinations and stress limits for Class 1 components and supports are given in Tables 3.9-2 and 3.9-3. Detail load combinations and stress limits for the pressurizer and safety and relief valve piping are described in Subsection 3.9.3.

3.9.3.1.2 ASME Code Class 2 and 3 Components and Supports

The loading combinations for ASME Code Class 2 and 3 components and supports furnished with the NPB are given in Table 3.9-5.

The allowable stress limits established for the components are sufficiently low to assure that violation of the pressure retaining boundary will not occur. These limits, for each of the loading combinations, are component oriented and are presented in Table 3.9-5. Active[a] pumps and valves are further discussed in paragraph 3.9.3.2. The component supports are designed in accordance with ASME B&PV Code, Section III, Subsection NF.

3.9.3.1.3 Analysis of Primary Components and Valves

Primary components that serve as part of the pressure boundary in the reactor coolant loop include the steam generators, reactor coolant pumps, pressurizer,

a. Active components are those whose operability is relied upon to perform a safety function (as well as reactor shut down function) during the transients or events considered in the respective operating condition categories.

piping, and reactor vessel. This equipment is American Nuclear Society (ANS)Safety Class 1 and the pressure boundary meets the requirements of ASME Code, Section III. This equipment is evaluated for the loading combinations outlined in Table 3.9-2. The equipment is analyzed for (1) the normal loads of weight, pressure, and temperature, (2) mechanical transients of OBE, SSE, and auxiliary line pipe ruptures, and (3) pressure and temperature transients outlined in Section 3.9.1.1.

The results of the reactor coolant loop analysis are used to determine the loads acting on the equipment nozzles and the support/component interface locations. These loads are supplied for all loading conditions on an "umbrella" load basis. That is, on the basis of previous plant analyses, a set of loads is determined which should be larger than those seen in any single plant analysis. The umbrella loads represent a conservative means of allowing detailed component analysis prior to the completion of the system analysis. Upon completion of the system analysis, conformance is demonstrated between the actual plant loads and the loads used in the analyses of the components. Any actual loads larger than the umbrella loads are evaluated by individualized analysis.

Seismic analyses are performed individually for the RCP, the pressurizer, and the steam generator. Detailed and complex dynamic models are used for the dynamic analyses. The response spectra corresponding to the building elevation at the highest component/building attachment elevation is used for the component analysis. Seismic analyses for the steam generator, RCP, and pressurizer are performed using 5 percent damping for the OBE and 8 percent damping for the SSE. The reactor pressure vessel is seismically qualified in accordance with ASME III by the reactor vessel vendor. The loadings used in the analysis are supplied by Westinghouse and are based on loads generated by a dynamic system analysis.

Auxiliary equipment that serves as part of the reactor coolant system pressure boundary include Class 1 valves and Class 1 auxiliary piping. Class 1 valves in the RCS are designed and analyzed according to the requirements of Subsection NB-3500 of ASME Code, Section III. This equipment is evaluated for the loading combinations and stress limits in Tables 3.9-2 and 3.9-3. The operability criteria for these valves are described in Section 3.9.3.2. Valves in sample lines connected to the RCS are not considered to be ANS Safety Class 1 nor ASME Class 1. This is because the nozzles where the lines connect to the primary system piping are orificed to a 3/8-inch hole. This hole restricts the flow so that loss through a severance of one of these lines can be made up by normal charging flow.

3.9.3.2 Pump and Valve Operability Assurance

3.9.3.2.1 Pumps

Safety related active pumps are subjected to in-shop tests which include hydrostatic tests of casing to 150 percent of the design pressure, and performance tests to determine the following:

- o Total developed head.
- o Minimum and maximum head.
- o Net positive suction head (NPSH) requirements except as noted below.
- o Other pump/motor characteristics

Where applicable, bearing temperature and vibration are monitored during the performance tests. After the pump is installed, it undergoes cold hydrostatic testing, hot functional testing, and applicable periodic inservice inspection and testing to verify and assure the functional ability and reliability of the pump for the design life of the plant.

In addition to the required testing, the pumps are designed and supplied in accordance with the following specified criteria:

A. In order to ensure that the active pump will not be damaged during the seismic event, the pump manufacturer must demonstrate by test or analysis that the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, will be considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis is performed. The natural frequency of the support is determined and used in conjunction with the project seismic response spectra. The deflection determined from the static shaft analysis is compared with the applicable clearances.

   If the natural frequency is found to be below 33 Hz, a dynamic analysis is performed using a finite element model to determine the amplified input accelerations necessary to perform the shaft analysis. The shaft deflection analyses are performed using the adjusted accelerations and the deflections compared with allowable shaft clearances. Assumptions used for generating the analytical model are verified by test.

B. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to ensure that unacceptable system misalignment cannot occur.

C. To complete the pump qualification, the pump motor and all appurtenances vital to the operation of the pump are independently qualified for operation within their specified environment, as well as during the maximum seismic event in accordance with Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1975.

   From this, it is concluded that the safety-related pump/motor assemblies will not be damaged, will continue operating under safe shutdown earthquake (SSE) loadings, and will perform their intended functions. These proposed requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

3.9.3.2.2 Valves

Safety-related, active valves are subjected to a series of stringest tests prior to service and during the plant life. Prior to installation, the following tests are performed: shell hydrostatic test, backseat and main seat leakage tests, disc hydrostatic tests, and operational tests to verify that the valve opens and closes. For the operability qualification of motor operators for the environmental conditions over the installed life, refer to Section 3.11 and Subsection 3.1.3, Regulatory Guide 1.73. Cold hydro tests, hot functional tests, periodic inservice inspections, and periodic inservice operations are performed in situ to verify and assure the functional ability of the valve. These tests guarantee reliability of the valve for the design life of the plant. The valves are constructed in accordance with the ASME B&PV Code, Section III. On active valves, an analysis of the extended structure is performed for static equivalent seismic SSE loads applied at the center of gravity of the extended structure. The stress limits used for active Class 2 and Class 3 valves are shown in Table 3.9-5. In addition to these tests and analyses, representative valves of each design type are tested for verification of operability during a simulated plant faulted condition event by demonstrating operational capabilities within the specified limits. The testing procedures are described below.

The valve is mounted in a manner that will conservatively represent typical valve installations. The valve includes the operator, accessory solenoid valves, and limit switches when attached to the valve in service. The operability of the valve during a faulted condition is demonstrated by satisfying the following criteria:

A. Active valves shall have a first natural frequency that is not less than 33 Hz.

B. A static load or loads equivalent to those resulting from the faulted condition accelerations is applied to the extended structure center of gravity so that the resulting deflection is in the nearest direction of the extended structure. The design pressure of the valve is applied to the valve during the static deflection tests.

C. The valve is cycled while in the deflected position. The valve must function within the specified operating time limits while subject to design pressure.

D. Electrical motor operators, limit switches, and pilot solenoid valves necessary for operation are qualified in accordance with IEEE Seismic Qualification Standards. IEEE Standard 344 is used for this qualification.

The above testing program applies to valves with extended structures. The testing is conducted on a representative number of valves. Valves from each of the primary safety related design types are tested. Valve sizes that cover the range of sizes in service are tested.

Valves that are safety related, but can be classified as not having an extended structure such as check valves and safety valves, are considered separately.

Check valves are characteristically simple in design, and their operation is not affected by seismic accelerations or the maximum applied nozzle loads. The check valve design is compact, and there are no extended structures or masses whose motion could cause distortions that could restrict operation of the valve. The design of these valves is such that once the structural integrity of the valve is assured, using standard design or analysis methods, the ability of the valve to operate is assured by the design features. In addition to these design considerations, the valve also undergoes the following: (1) in-shop hydrostatic test, (2) in-shop seat leakage test, and (3) periodic in situ valve exercising and inspection.

Pressurizer safety valves are qualified for operability in the same manner as valves with extended structures, as described above. The qualification methods include analysis of the bonnet for static equivalent SSE loads, in shop hydrostatic and seat leakage tests, and periodic in situ valve inspection. Additionally, representative pressurizer safety valves are tested to verify analysis methods. This test is described as follows:

A. The safety valve is mounted to represent the specified installation.

B. The valve body is pressurized to its normal system pressure.

C. A static load representing the faulted condition seismic load is applied to the top of the valve bonnet in the weakest direction of the extended structure.

D. The pressure is increased until the valve actuates.

E. Actuation of the valve at its setpoint ensures its operability during the faulted condition.

Using these methods, all the safety related valves in the system are qualified for operability during a faulted event. These methods conservatively simulate the seismic event, and assure that the active valves perform their safety related function when necessary.

3.9.3.2.3 Pump Motor and Valve Operator Qualification

Active pump motors and active valve motor operators, limit switches, and solenoid valves are seismically qualified in accordance with IEEE Standard 344-1975.

3.9.3.2.4 Active ASME Code Class 2 and 3 Pumps

Safety related active pumps are subjected to in-shop tests that include hydrostatic tests of casing to 150 percent of the design pressure, and performance tests to determine total developed head, minimum and maximum head, net positive suction head (NPSH) requirements, and other pump/motor characteristics. Vibration is monitored during the performance tests.

In addition to the required testing, the pumps are designed and supplied in accordance with the following specified criteria:

A. In order to ensure that the active pump will not be damaged during the seismic event, the pump manufacturer is required to demonstrate by test or analysis that the lowest natural frequency of the pump is greater than 33 Hz. The pump, when having a natural frequency above 33 Hz, will be considered essentially rigid. This frequency is considered sufficiently high to avoid problems with amplification between the component and structure for all seismic areas. A static shaft deflection analysis of the rotor is performed. The natural frequency of the support is determined and used in conjunction with the project seismic response spectra. The deflection determined from the static shaft analysis is compared to the allowable rotor clearances.

   If the natural frequency is found to be below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis. The static deflection analyses are performed using the adjusted accelerations.

B. The maximum seismic nozzle loads are also considered in an analysis of the pump supports to ensure that unacceptable system misalignment cannot occur.

C. To complete the seismic qualification procedures, the pump motor and all appurtenances vital to the operation of the pump are independently qualified for operation during the maximum seismic event in accordance with IEEE Standard 344-1975.

From this, it is concluded that the safety related pump/motor assemblies will not be damaged and will continue operating under SSE loadings and will perform their intended functions. These proposed requirements take into account the complex characteristics of the pump and are sufficient to demonstrate and assure the seismic operability of the active pumps.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

3.9.3.3.1 Pressure Relief Devices on NPB Components

The pressurizer safety and relief valve (PSARV) discharge piping systems provide overpressure protection for the RCS. The spring-loaded safety valves located on top of the pressurizer are designed to prevent system pressure from exceeding design pressure by more than ten percent. The power-operated relief valves, also located on top of the pressurizer, are designed to prevent system pressure from exceeding the normal operating pressure by more than 100 psi. A water seal is maintained upstream of each valve to minimize leakage. Condensate accumulation on the inlet side of each valve prevents any leakage of hydrogen gas or steam through the valves. The valve outlet side is sloped to prevent the formulation of additional water pockets.

If the pressure exceeds the setpoint and the valve opens, the water slug from the loop seal discharges. The water slug, driven by high system pressure, generates transient thrust forces at each location where a change in flow direction or area occurs. The valve discharge conditions considered in the thrust analysis of the PSARV piping systems are as follows: 1) the safety valves are assumed to open simultaneously while the relief valves remain closed, and 2) the relief valves open simultaneously while the safety valves are closed.

In addition to these two cases, which consider water seal discharge (water slug) followed by steam, solid water from the pressurizer (cold overpressure) is also analyzed.

For each pressurizer safety and relief piping system, an analytical hydraulic model is developed. The piping from the pressurizer nozzle to the relief tank nozzle is modeled as a series of single pipes. The pressurizer is modeled as a reservoir which contains steam at constant pressure (2500 psia for safety system and 2350 psia for relief system) and at constant temperature of 680°F. The pressurizer relief tank is modeled as a sink which contains steam and water mixture.

Fluid acceleration inside the pipe generates reaction forces on all segments of the line which are bounded at either end by an elbow or bend. Reaction forces resulting from fluid pressure and momentum variations are calculated. These forces are defined in terms of the fluid properties for the transient hydraulic analysis.

Unbalanced forces are calculated for each straight segment of pipe from the pressurizer to the relief tank. The hydraulic analysis includes the effect of water slug discharge. The time histories of these forces are used for the subsequent structural analysis of the pressurizer safety and relief lines.

The structural model used in the seismic analysis of the safety and relief lines is modified for the valve thrust analysis to represent the safety and relief valve discharge. The time-history hydraulic forces are applied to the piping system lump mass points. The dynamic solution for the valve thrust is obtained by using a modified predictor-corrector-integration technique and normal mode theory.

The time-history solution is performed in subprogram FIXFM. The input to this subprogram consists of the natural frequencies and normal modes, applied forces, and nonlinear elements. The natural frequencies and normal modes for the modified pressurizer safety and relief line dynamic model are determined with the WESTDYN program. The support loads are computed by multiplying the support stiffness matrix and the displacement vector at each support point. The time-history displacements of the FIXFM subprogram are used as input to the WESDYN2 subprogram to determine the internal forces, deflections, and stresses at each end of the piping elements.

The loading combinations considered in the analysis of the PSARV piping are given in Tables 3.9-6 through 3.9-9.

3.9.3.3.2 Other Pressure Relief Devices on Components

The design of pressure-relieving devices can be generally grouped in two categories: open discharge and closed discharge.

A. Open Discharge:

An open discharge is characterized by a relief or safety valve discharging to the atmosphere or to a vent stack open to the atmosphere.

The design of open discharge valve stations includes the following considerations:

1. Stresses in the valve header, the valve inlet piping, and local stresses in the header-to-valve inlet piping junction due to thermal effects, internal pressure, seismic loads, and thrust loads are considered. These stresses are calculated in accordance with the applicable subsections of Section III of the ASME B&PV Code.

2. Thrust forces include both pressure and momentum effects.

3. Where more than one safety or relief valve is installed on the same pipe run, valve spacing is as specified in ASME Code.

4. Where more than one safety or relief valve is installed on the same pipe run, the sequence of openings that induces the maximum stresses is considered as recommended by Regulatory Guide 1.67.

5. The minimum moments to be used in stress calculations are those specified in ASME Code.

6. The effects of the valve discharge on piping connected to the valve header are considered.

7. The reaction forces and moments used in stress calculations include the effects of a dynamic load factor (DLF), or are the maximum instantaneous values obtained from a time-history structural analysis. A dynamic load factor of 2.0 is used, if a dynamic structural analysis is not performed, to determine the dynamic load factor as recommended by Regulatory Guide 1.67.

B. Closed Discharge

A closed discharge system is characterized by piping between the valve and a tank or some other terminal end. Under steady-state conditions, there are no net unbalanced forces. The initial transient response and resulting stresses are determined using either a time-history computer solution or a conservative equivalent static solution. In calculating initial transient forces, pressure and momentum terms are included. Water slug effects are also included.

3.9.3.4 Component and Piping Supports

For statically applied loads, the stress allowables of Appendix F of ASME Code, Section III are used for Code components.

Dynamic loads for components loaded in the elastic range are calculated using dynamic load factors, time-history analysis, or any other method that assumes elastic behavior of the component. A component is assumed to be in the elastic range if yielding across a section does not occur. The limits of the elastic range are defined in Paragraph F-1323 of Appendix F for Code components. Local yielding due to stress concentration is assumed not to affect the validity of the assumptions of elastic behavior. The stress allowables of Appendix F for elastically analyzed components are used for Code components.

For non-Code components, allowables are based on tests or accepted industry standards comparable to those in Appendix F of ASME Code, Section III.

3.9.3.4.1 ASME Code Class 1 Component Supports

The load combinations and allowable stresses for ASME Code Class 1 components and component supports are given in Tables 3.9-2 and 3.9-3, respectively.

3.9.3.4.1.1 Primary Component Supports Models and Methods

The static and dynamic structural analyses employed the matrix method and normal mode theory for the solution of lumped-parameter, multimass structural models. The equipment support structure models are dual-purpose, since they are required to represent quantitatively the elastic restraints that the supports impose upon the loop, and to evaluate the individual support member stresses due to the forces imposed upon the supports by the loop.

A description of the supports is found in Subsection 5.4.14 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System." Detailed models of the supports are developed using beam elements and plate elements, where applicable. The reactor vessel supports are modeled using the WECAN computer program. Steam generator and RCP supports are normally modeled as linear or nonlinear springs.

For each operating condition, the loads (obtained from the reactor coolant loop analysis) acting on the support structures are appropriately combined. The adequacy of each member of the steam generator supports, RCP supports, and piping restraints for auxiliary connections is verified by solving the ASME Code, Section III, Subsection NF stress and interaction equations. The adequacy of the reactor pressure vessel support structure is verified using the WECAN computer program and comparing the resultant stresses to the criteria given in ASME Code, Section III, Subsection NF.

The test load method given in F-1370(d) is an acceptable method of qualifying components in lieu of satisfying the stress/load limits established for the component analysis.

The test load option is used to qualify the reactor pressure vessel nozzle support pads. To duplicate the loads that act on the pads during faulted conditions, the tests, which utilized a one-eighth linear scale model, are performed by applying a unidirectional static load to the nozzle pad. The load on the nozzle pad was reacted upon by a support shoe that was mounted to the test fixture.

The modeling and application of the load thus allows the maximum load capacity of the support pads to be established accurately. The test load, L_T, is then determined by multiplying the maximum collapse load by sixty-four (ratio of prototype area to model area), and included temperature effects in accordance with the rules of the ASME Code, Section III.

The loads on the reactor vessel support pads, as calculated in the system analysis for faulted conditions, are limited to the value of .80 L_T. The tests performed and the limits established for test load method ensure that the experimentally obtained value for L_T is accurate and that the support pad design is adequate for its intended function.

3.9.3.4.2 ASME Code Class 2 and 3 Supports

Class 2 and 3 component supports are designed and analyzed for design, normal, upset, emergency, and faulted conditions to the rules and requirements of Section III, Subsection NF of the ASME Code. The stress limits for Class 2 and 3 component supports for all loading conditions are defined in Table 3.9-5. The analyses or test methods and associated stress or load allowable limits that are used in the evaluation of linear supports for faulted conditions are those defined in Appendix F of the ASME Code. Plate and shell type supports satisfy the faulted condition limits provided in Subsection NF, Paragraph 3321, of the ASME Code, Section III. Supplementary requirements are presented in Subsection 3.9.3.2.1 which include stress analysis and evaluation of pump/motor support alignment. Thus, the operability of active pumps is not compromised by the supports during faulted conditions. The allowable stresses and loading combinations for ASME Code Class 2 and 3 component and component supports are given in Tables 3.9-4 and 3.9-5.

3.9.3.4.3 Snubbers Used as Component Supports

The location and size of the snubbers are determined by stress analysis. The location and line of action of a snubber are selected based on the necessity of limiting seismic stresses in the piping and nozzle loads on equipment. Snubbers are chosen in lieu of rigid supports where restricting thermal growth would induce excessive thermal stresses in the piping or nozzle loads or equipment. The snubbers are constructed to ASME Boiler and Pressure Vessel Code, Section III, Subsection NF standards.

Two types of tests are performed on the snubber.

A. Production tests are made on every unit:

B. Qualification tests are performed on randomly selected production models to demonstrate the required load performance (load rating).

In the piping system seismic stress analysis, the mechanical snubbers are modeled as stops. Where necessary, the snubber spring rates are incorporated into the analysis.

The recommendations of Regulatory Guide 1.124 applicable to the service limits and loading combinations for Class 1 linear supports are met as discussed in Table 3.9-5.

A tabulation of snubbers utilized as supports for safety-related systems and components is provided in the Technical Specifications.

Supports for active pumps and valves are included in the overall design and qualification of the component.

Design specifications for snubbers include:

- Seismic requirements.
- Normal environmental parameters.

- o Accident/post-accident environmental parameters.

- o Full-scale performance test to measure pertinent performance requirements.

- o Instructions for periodic maintenance (in technical manuals).

3.9.4 Control Rod Drive Systems

Descriptive information on the control rod drive mechanism (CRDM) and gray rod drive mechanism (GRDM) is provided in Section 3.9.4 of RESAR-SP/90 PDA Module 5, "Reactor System."

Information relating to design specifications and design stresses for the drive mechanisms is provided in Sections 3.1 and 3.9.3 of this module, and 4.5 of RESAR-SP/90 PDA Module 5, "Reactor System."

The control rod drive mechanisms (CRDMs), the gray rod drive mechanisms (GRDM) and their support structures are evaluated for the loading combinations outlined in Table 3.9-2.

A detailed finite-element model of the drive mechanisms and supports is constructed using the WECAN computer program with beam, pipe, and spring elements. For the LOCA analysis, nonlinearities in the structure are represented. These include RPI plate impact, tie rods, and lifting leg clevis/RPV head interface. The time-history motion of the reactor vessel head, obtained from the RPV analysis, is input to the dynamic model. Maximum forces and moments in the drive mechanisms and support structures are then determined. For the seismic analysis, the structural model is linearized and the floor response spectra corresponding to the drive mechanisms tie rod elevation is applied to determine the maximum forces and moments in the structure.

The bending moments calculated for the drive mechanisms for the various loading conditions are compared with maximum allowable moments determined from a detailed finite-element stress evaluation of the drive mechanisms. Adequacy of the drive mechanisms support structure is verified by comparing the calculated stresses to the criteria given in ASME Code, Section III, Subsection NF.

Operational transients are listed in Section 3.9.1 of this module.

3.9.5 Reactor Pressure Vessel Internals

Information on the design arrangements, loading conditions, and design bases is provided in Section 3.9.5 of RESAR-SP/90 PDA Module 5, "Reactor System."

The structural analysis of reactor vessel and internals consider simultaneous application of the time-history loads resulting from the reactor coolant loop mechanical loads and internal hydraulic pressure transients. The vessel is restrained by reactor vessel supports at the reactor vessel nozzles, and by the reactor coolant loops with the primary supports of the steam generators and the RCPs.

3.9.5.1 Loading Conditions

Following a postulated auxiliary line pipe rupture, the reactor vessel is excited by time-history forces. As previously mentioned, these forces are the combined effect of phenomena: (1) reactor coolant loop mechanical loads and (2) reactor internals hydraulic forces.

The reactor coolant loop mechanical forces are derived from the elastic analysis of the loop piping for the postulated auxiliary line break. The reactions on the nozzles of all the unbroken piping legs are applied to the vessel in the reactor pressure vessel blowdown analysis.

The internals reaction forces develop from asymmetric pressure distributions inside the reactor vessel. For an auxiliary line break on the vessel inlet leg, the depressurization wave path is through the broken loop inlet nozzle and into the region between the core barrel and reactor vessel. This region is called the downcomer annulus. The initial waves propagate up, down and around the downcomer annulus and up through the fuel. In the case of an auxiliary line break on the vessel outlet leg, the wave passes through the reactor pressure vessel outlet nozzle and directly into the upper internals region, depressurizes the core, and enters the downcomer annulus from the bottom of the vessel.

Thus, for an outlet leg auxiliary line break, the downcomer annulus is depressurized with much smaller differences in pressure horizontally across the core barrel than for the inlet leg auxiliary line break. For both breaks, the depressurization waves continue their propagation by reflection and translation through the reactor vessel fluid but the initial depressurization wave has the greatest effect on the loads.

The reactor internals hydraulic pressure transients are calculated including the assumption that the structural motion is coupled with the pressure transients. This phenomena has been referred to as hydro-elastic coupling or fluid-structure interaction. The hydraulic analysis considers the fluid-structure interaction of the core barrel by accounting for the deflections of constraining boundaries which are represented by masses and springs. The dynamic response of the core barrel in its beam bending mode responding to blowdown forces compensates for internal pressure variation by increasing the volume of the more highly pressurized regions.

3.9.5.2 Reactor Vessel and Internals Modeling

The mathematical model of the reactor pressure vessel is a three-dimensional nonlinear finite-element model which represents the dynamic characteristics of the reactor vessel and its internals in the six geometric degrees of freedom. The model was developed using the WECAN computer code. The model consists of three concentric structural submodels connected by nonlinear impact elements and stiffness matrices. The first submodel represents the reactor vessel shell and associated components. The reactor vessel is restrained by the reactor vessel supports and by the attached primary coolant piping. Each reactor vessel support is modeled by a linear horizontal stiffness and a vertical impact element. The attached piping is represented by a stiffness matrix.

The second submodel represents the reactor core barrel, neutron panels, lower support plate, tie plates, and secondary core support components. This submodel is physically located inside the first, and is connected to it by a stiffness matrix at the internals support ledge. Core-barrel-to-vessel-shell impact is represented by nonlinear elements at the core barrel flange, core barrel nozzle, and lower radial support locations.

The third and innermost submodel represents the upper support plate, guide tubes, support columns, upper and lower core plates, and fuel. The third submodel is connected to the first and second by linear stiffness and nonlinear elements.

3.9.5.3 Analytical Methods

The time-history effects of the internals loads and loop mechanical loads are combined and applied simultaneously to the appropriate nodes of the mathematical model of the reactor vessel and internals. The analysis is performed by numerically integrating the differential equations of motion to obtain the transient response. The output of the analysis includes the displacements of the reactor vessel and the loads in the reactor vessel supports which are combined with other applicable faulted condition loads and subsequently used to calculate the stresses in the supports. Also, the reactor vessel displacements are applied as a time-history input to the dynamic reactor coolant loop blowdown analysis. The resulting loads and stresses in the piping components and supports include both loop blowdown loads and reactor vessel displacements. Thus, the effect of the vessel displacements upon loop response and the effect of loop blowdown upon vessel displacements are both evaluated.

3.9.6 Inservice Testing of Pumps and Valves

Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda, as required by 10 CFR 50, Section 50.55a(g).

A preservice inspection program (nondestructive examination) and a preservice test program (pumps and valves) for each unit will be prepared. The inservice inspection program and inservice test program will be prepared within 6 months after the units' operating license issue date. These programs will comply with applicable inservice inspection provisions of 10 CFR 50.55a(g). The preservice programs will provide details of areas subject to examination, as well as the method and extent of preservice examinations. Inservice programs will detail the areas subject to examination and method, extent, and frequency of examinations after start-up.

3.9.6.1 Inservice Testing of Pumps

The pump test program will list all safety-related Class 1, 2, and 3 pumps that are provided with an emergency power source and are necessary to shut the plant down safely or mitigate the consequences of an accident. The pump test program will be in accordance with Subsection IWP of the ASME Code, Section XI, to the extent practical, and comply with all applicable portions of 10 CFR 50.55a(g). The hydraulic and mechanical test parameters to be measured or observed will be defined in a separate inservice inspection program.

3.9.6.2 Inservice Testing of Valves

The valve test program will list all safety-related (i.e., those valves necessary to shut the plant down safely or mitigate the consequences of an accident) Class 1, 2, and 3 valves subject to operational readiness testing and will indicate the test parameters to be measured or observed. The test program will conform to the requirements of ASME Code, Section XI, Subsection IWV, to the extent practical, and comply with all applicable portions of 10 CFR 50.55a(g). Test parameters to be measured or observed will be defined in a separate inservice inspection program.

3.9.6.3 Relief Requests

Relief from the testing requirements of Section XI will be requested when full compliance with requirements of the code is not practical. In such cases, specific information will be provided which identifies the applicable code requirements, justification for the relief request, and the testing method to be used as an alternative.

Table 3.9-1

(Sheet 1 of 4)

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

Level A Service Conditions (Normal Conditions)	Occurrences
A. Reactor coolant pump start-up/shutdown	4,000
B. Heatup and cooldown	200 (each)
C. Unit loading and unloading between 0 and 15% of full power	500 (each)
D. Unit loading and unloading between 15 and 100% of full power	13,200 (each)
E. Reduced temperature return to power	2,000
F. Step load increase and decrease of 10% of full power	2,000 (each)
G. Large step load decrease with steam dump	200
H. Load Regulation	See discussion
I. Boron concentration equalization	13,200
J. Feedwater cycling	2,000
K. Loop out of service	
a. Normal loop shutdown	40
b. Normal loop start-up	30

Table 3.9-1

(Sheet 2 of 4)

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

		Occurrences
L.	Refueling	80
M.	Turbine roll test	20
N.	Primary side leakage test	200
O.	Secondary side leakage test	80
P.	Core life time extension	40
Q.	Feedwater heaters out of service	120
Level B Service Conditions (Upset Conditions)		
A.	Loss of load	40
B.	Loss of offsite power	40
C.	Partial loss of flow	40
D.	Reactor trip from low power	200
E.	Reactor trip from full power	
	a. With no cooldown	80
	b. With cooldown and no safety injection	80
	c. With cooldown and safety injection	40

Table 3.9-1

(Sheet 3 of 4)

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

		Occurrences
F.	Inadvertent reactor coolant system depressurization	10
G.	Inadvertent start-up of an inactive loop	10
H.	Control rod drop	40
I.	Excessive feedwater flow	30
J.	Cold overpressurization	10
K.	Sudden stoppage of flow	5
L.	Operating basis earthquake	50 cycles
Level C Service Conditions (Emergency Conditions)[a]		
A.	Small loss-of-coolant accident	5
B.	Small steam break	5
C.	Small feedwater line break	5

(a) In accordance with the ASME Code Section III, Nuclear Power Plant Components, emergency and faulted conditions are not included in fatigue evaluations.

Table 3.9-1

(Sheet 4 of 4)

SUMMARY OF REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

Level D Service Conditions (Faulted Conditions)[a]	Occurrences
A. Primary Coolant System Auxiliary Line Pipe Break (Large Loss-of-Coolant Accident)	1
B. Large steam break	1
C. Feedwater line break	1
D. Reactor coolant pump locked rotor	1
E. Control rod ejection	1
F. Steam generator tube rupture	1
G. Safe shutdown earthquake	1
Test Conditions	
A. Primary side hydrostatic test	10
B. Secondary side hydrostatic test	10
C. Tube leakage test	800

(a) In accordance with the ASME Code Section III, Nuclear Power Plant Components, emergency and faulted conditions are not included in fatigue evaluations.

Table 3.9-2

LOADING COMBINATIONS FOR ASME CLASS 1
COMPONENTS AND SUPPORTS

Plant Classification	Design/Service Level	Loading Combination
Design	Design	Design pressure, design temperature, deadweight
Normal	Service level A	Normal condition transients, deadweight
Upset	Service level B	Upset condition transients, deadweight, OBE
Emergency	Service level C	Emergency condition transients, deadweight
Faulted	Service level D	Faulted condition transients, deadweight, SSE, pipe rupture loads

Table 3.9-3
(Sheet 1 of 2)

STRESS CRITERIA FOR ASME B&PV CODE SECTION III
CLASS 1 COMPONENTS[a] AND SUPPORTS

Design/Service Level	Vessels/Tanks	Piping	Pumps	Valves	Component Supports[c][d]
Design and service level A	ASME B&PV Code, Section III NB-3221, 3222	ASME B&PV Code, Section III NB-3652, 3653	ASME B&PV Code, Section III NB-3221, 3222	ASME B&PV Code, Section III NB-3520, 3525	ASME B&PV Code, Section III Subsection NF NF-3221, 3222 NF-3231.1(a) NF-3240
Service level B (Upset)	ASME B&PV Code, Section III NB-3223	ASME B&PV Code, Section III NB-3654	ASME B&PV Code, Section III NB-3223	ASME B&PV Code, Section III NB-3525	ASME B&PV Code, Section III Subsection NF NF-3223, 3231.1(a) NF-3240
Service level C (Emergency)	ASME B&PV Code, Section III NB-3224	ASME B&PV Code, Section III NB-3655	ASME B&PV Code, Section III NB-3224	ASME B&PV Code, Section III NB-3526	ASME B&PV Code, Section III Subsection NF NF-3224, 3231.1(b) NF-3240
Service level D (Faulted)	ASME B&PV Code, Section III see paragraph 3.9.1.4 NB-3225	ASME B&PV Code, Section III see paragraph 3.9.1.4 NB-3656	ASME B&PV Code, Section III (No active Class 1 pump used) NB-3225	(b)	ASME B&PV Code, Section III Subsection NF, see paragraph 3.9.1 NF-3225, 3231.1(c) NF-3240

P_m, P_b, P_s, P_{ed}, Q_{t2}, S_u, C_p, S_n and S_m as defined by ASME B&PV Code, Section III

Table 3.9-3
(Sheet 2 of 2)

STRESS CRITERIA FOR ASME B&PV CODE SECTION III
CLASS 1 COMPONENTS(a) AND SUPPORTS

CLASS I VALVE SERVICE LEVEL D CRITERIA

a. A test of the components may be performed in lieu of analysis.

ACTIVE	INACTIVE
Calculate Pm from Subsection NB-3545.1 with Internal Pressure Ps = 1.25 Ps Pm $\leq$ 1.5 Sm	Calculate Pm from Subsection NB-3545.1 with Internal Pressure Ps = 1.50 Ps Pm $\leq$ 2.45 Sm or 0.7 Su
Calculate Sn from Subsection NB-3545.2 with Cp = 1.5 Ps = 1.25 Ps Qt2 = 0 Ped = 1.3X value of Ped from equations of NB-3545.2 Sn $\leq$ 3 Sm	Calculate Sn from Subsection NB-3545.2 with Cp = 1.5 Ps = 1.50 Ps Qt2 = 0 Ped = 1.3X value of Ped from equations of NB-3545.2 Sn $\leq$ 3 Sm

c. Including pipe supports.

d. In instances where the determination of allowable stress values utilizes S_u (ultimate tensile stress) at temperatures not included in ASME III, S_u shall be calculated using one of the methods provided in Regulatory Guide 1.124, Revision 1.

Table 3.9-4

LOADING COMBINATIONS FOR ASME CODE CLASS 2 AND 3
COMPONENTS AND SUPPORTS FOR THE NPB

Plant Condition	Design/Service Level Requirements	Loading Combination[a,b]
Design	Design	Design pressure, Design temperature, Deadweight
Normal	Service Level A	Normal condition pressure, normal condition metal temperature, deadweight
Upset	Service Level B	Upset condition pressure, upset condition metal temperature, deadweight, OBE
Emergency	Service Level C	Emergency condition pressure, emergency condition metal temperature, deadweight
Faulted	Service Level D	Faulted condition pressure, faulted condition metal temperature, deadweight, SSE, pipe rupture

a. Temperature is used to determine allowable stress only.

b. Pressure, and temperatures are those associated with the respective plant conditions (i.e., normal, upset, emergency, and faulted), as noted, for the component under consideration.

Table 3.9-5

STRESS CRITERIA FOR ASME B&PV CODE SECTION III
CLASS 2 AND 3 COMPONENTS

Design/Service Level	Vessels/Tanks	Piping	Pumps	Valves	Component Supports
Design and Service Level A	ASME B&PV Code Section III NC 3217 NC/ND-3310, 3320	ASME B&PV Code Section III NC/ND-3652, 3653	ASME B&PV Code Section III NC/ND-3400	ASME B&PV Code Section III NC/ND-3510	ASME B&PV Code Section III NF-3321 NF-3231 NF-3260
Service Level B (upset)	ASME B&PV Code Section III NC/ND-3310, 3320	ASME B&PV Code Section III NC/ND-3653	ASME B&PV Code Section III NC/ND-3400	ASME B&PV Code Section III NC/ND-3520	ASME B&PV Code Section III NF-3321 NF-3231 NF-3260
Service Level C (Emergency)	ASME B&PV Code Section III NC/ND-3310, 3320	ASME B&PV Code Section III NC-3654	ASME B&PV Code Section III NC/ND-3400	ASME B&PV Code Section III NC/ND-3520	ASME B&PV Code Section III NF-3321 NF-3231 NF-3260
Service Level D (Faulted)	ASME B&PV Code Section III NC/ND-3310, 3320	ASME B&PV Code Section III NC-3655	ASME B&PV Code Section III NC/ND-3400	ASME B&PV Code Section III NC/ND-3520	ASME B&PV Code Section III NF-3321 NF-3231 NF-3260

Table 3.9-6

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER
SAFETY AND RELIEF VALVE PIPING - UPSTREAM OF VALVES

Combination	Plant/System Operating Condition	Load Combination	Piping Allowable Stress Intensity
1	Normal	N	1.5 S_m
2	Upset	N + OBE + SOT_U	1.8 S_m/1.5 S_y
3	Emergency	N + SOT_E	2.25 S_m/1.8 S_y
4	Faulted	N + SSE + SOT_F	3.0 S_m

1. Table 3.9-8 contains SOT definitions and other load abbreviations.
2. SRSS is to be used for combining dynamic load responses.
3. This also applies to pressurizer nozzles and valve support brackets.

Table 3.9-7

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR
PRESSURIZER SAFETY AND RELIEF VALVE PIPING
SEISMICALLY DESIGNED DOWNSTREAM PORTION[1][2]

Combination	Plant/System Operating Condition	Load Combination	Piping Allowable Stress Intensity
1	Normal	N	1.0 S_h
2	Upset	N + SOT_U	1.2 S_h
3	Upset	N + OBE + SOT_U	1.8 S_h
4	Emergency	N + SOT_E	1.8 S_h
5	Faulted	N + SSE + SOT_F	2.4 S_h

1. Table 3.9-8 contains SOT definitions and other load abbreviations.
2. SRSS is to be used for combining dynamic load responses.

Table 3.9-8

DEFINITIONS OF LOAD ABBREVIATIONS

N	=	Sustained loads during normal plant operation
SOT	=	System operating transient
SOT_U	=	Relief valve discharge transient
SOT_E	=	Safety valve discharge transient
SOT_F	=	Max (SOT_U, SOT_E), or transition flow
OBE	=	Operating basis earthquake
SSE	=	Safe shutdown earthquake
S_h	=	Basic material allowable stress at maximum (hot) temperature
S_m	=	Allowable design stress intensity
S_y	=	Yield strength value

Table 3.9-9

LOAD COMBINATIONS FOR PRESSURIZERS SAFETY
AND RELIEF VALVE NOZZLES AND SUPPORT BRACKETS

ASME Code Section III Condition	External Load Combinations	Internal Pressure
Design I	DW + OBE	Design
Design II	$DW + VO_R$	Design
Normal/Upset I	DW + T + OBE	Transient
Normal/Upset II	$DW + T + VO_R$	Transient
Normal/Upset III	$DW + T + VO_S$	Transient
Emergency I	$DW + VO_S$	Transient
Emergency II	$DW + (VO_R^2 + OBE^2)^{1/2}$	Transient
Faulted I	$DW + (VO_S^2 + SSE^2)^{1/2}$	Transient

Table 3.9-10

PUMP STARTING/STOPPING CONDITIONS

Plant Condition	RCS (°F)/(psig)	SG Secondary (°F)/(psig)	Number of Starts/Stops	Operation
Cold	70/400	70/0	800	RCS venting
Cold	70/400	70/0	200	RCS heatup, cooldown
Restart	100/400	100/0	500	Hot functionals RCP stops, starts
Hot	567/2235	567/1183	1250	Transients and miscellaneous
Hot	567/2235	567/1183	1250	Transients and miscellaneous

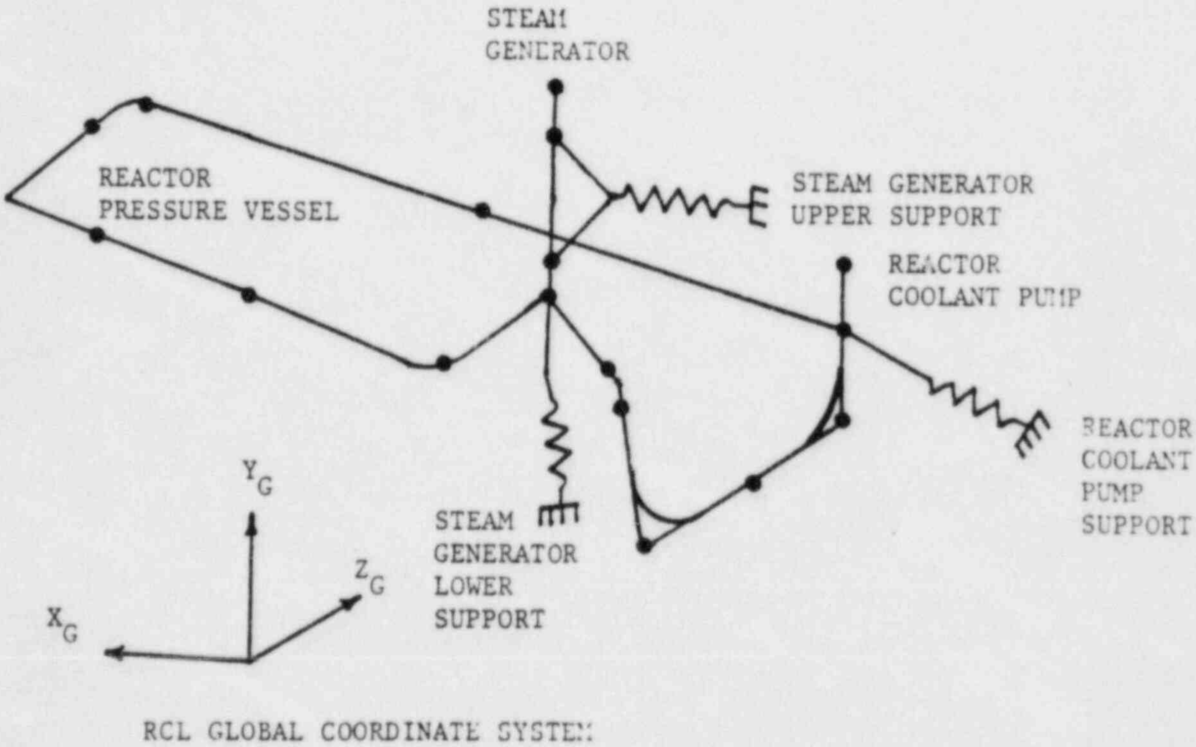

Figure 3.9-1. Reactor Coolant Loop Supports System, Dynamic Structural Model

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

This section presents information to demonstrate that instrumentation and electrical equipment in the single unit Nuclear Power Block (NPB), classified as Seismic Category I, are capable of performing designated safety related functions in the event of an earthquake. Items in the NPB scope include; (1) the containment building, (2) the fuel handling facilities, (3) the mechanical safeguards equipment area, (4) the auxiliary systems area, (5) the instrumentation and controls area, (6) the control room, (7) the electrical power distribution equipment area, (8) the emergency diesel generator area, and (9) the technical support center. A detailed listing of NPB scope items can be found in Table 1.1-1. The information presented includes identification of the Category I instrumentation and electrical equipment, the qualification criteria employed for each item of equipment, and documentation of the qualification process employed to demonstrate the required seismic capability.

3.10.1 Seismic Qualification Criteria

The Seismic Category I instrumentation and electrical equipment are designed to withstand the effects of the safe shutdown earthquake (SSE) and remain functional during and after accident conditions. The parameters used to develop seismic loadings and criteria for Seismic Category I instrumentation and electrical equipment are described in Section 3.7.

The Seismic Category I instrumentation and electrical equipment is divided into two classifications: that equipment which is designed to maintain its functional capability during and after an SSE, and equipment which, although not required to maintain its functional capability, is designed to maintain the pressure boundary integrity of the system of which it is a part during and after an SSE.

The performance requirements for the Seismic Category I electrical and instrumentation items and their respective supports are structural as well as functional. Where applicable, the structural requirements are in accordance with American Institute of Steel Construction (AISC), "Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings," adopted February 12, 1969, or similar codes applicable for other construction materials.

The structural requirements for instrumentation equipment and systems which are required to maintain pressure boundary integrity are in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1977 edition through 1977 Winter addenda.

The reactor protection system and engineered safety features actuation system (ESFAS) are designed with the capability to initiate a protective action during and after the SSE.

The NRC recommendations concerning the methods to be employed for seismic qualification of electrical equipment are contained in Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants," which endorses IEEE Standard 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." WAPWR will meet this standard, as modified by Regulatory Guide 1.100, by either type test, analysis, or an appropriate combination of these methods. WAPWR will meet this commitment employing the methodology described in the final NRC approved version of Reference 1.

3.10.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

In accordance with IEEE Standard 344-1975, seismic qualification of safety-related electrical equipment is demonstrated by either type testing, analysis, or a combination of these methods. The choice of qualification methods is based upon many factors including: practicability, complexity of equipment, economics, availability of previous seismic qualification to earlier standards, etc. The qualification method employed for a particular item of equipment is identified in the individual Equipment Qualification Data Packages (EQDP) of Reference 2.

3.10.2.1 Seismic Qualification by Type Test

From 1969 to mid-1974 Westinghouse seismic test procedures employed single axis sine beat inputs in accordance with IEEE Standard 344-1971, "IEEE Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations," to seismically qualify equipment. The input form selected by Westinghouse was chosen following an investigation of building responses to seismic events as reported in Reference 3. In addition, Westinghouse has conducted seismic retesting of certain items of equipment as part of the Supplemental Qualification Program (Reference 4). This retesting was performed at the request of the NRC on agreed selected items of equipment employing multifrequency, multiaxis test inputs (Reference 5) to demonstrate the conservatism of the original sine beat test method with respect to the modified methods of testing for complex equipment recommended by IEEE Standard 344-1975.

The original single axis sine beat testing and the additional retesting completed under the Supplemental Test Program has been the subject of generic review by the NRC. For equipment which has been previously qualified by the single axis sine beat method, included in the NRC seismic audit, and (where required by the NRC) included in the Supplemental Qualification Program (Reference 4), no additional qualification testing is required to demonstrate acceptability to IEEE Standard 344-1975 provided that:

1. The Westinghouse aging evaluation program for aging effects on complex electronic equipment located outside containment demonstrates there are no deleterious aging phenomena. In the event that the aging evaluation program identifies materials that are marginal, either the materials will be replaced or the projected qualified life will be adjusted.

2. Any changes made to the equipment due to item 1 above or due to design modifications do not significantly affect the seismic characteristics of the equipment.

3. The previously employed test inputs can be shown to be conservative with respect to applicable plant specific response spectra.

This equipment is identified in Reference 1 (Table 7.1) and the test results are provided in the applicable EQDPs of Reference 2.

For equipment tests after July, 1974 (i.e., new designs, equipment not previously qualified, or previously qualified equipment that does not meet items 1, 2, and 3 above) seismic qualification by test is performed in accordance with IEEE Standard 344-1975. Where testing is utilized, multifrequency multiaxis inputs are developed by the general procedures outlined in Reference 5. The test results contained in the individual EQDPs of Reference 2 demonstrate that the measured test response spectrum envelops the applicable required response spectrum defined for generic testing as specified in Section 1 of the EQDP (Reference 2). Qualification for plant specific use is established by verification that the generic required response spectrum specified by Westinghouse envelops the applicable plant specific response spectrum. Alternative test methods, such as single frequency or single axis inputs, are used in selected cases as permitted by IEEE Standard 344-1975 and Regulatory Guide 1.100.

3.10.2.2 Seismic Qualification by Analysis

Analysis without testing may be acceptable only if structural integrity alone can assure the design-intended function. The procedures described in Sections 5.2 through 5.4 of IEEE Standard 344-1975 are followed when analysis is used. The analysis is performed by the equipment supplier or a qualified consultant.

The structural integrity of safety related motors (see Table 3.10-1, EQDP AE-2 and AE-3) is demonstrated by a static seismic analysis in accordance with IEEE Standard 344-1975, with justification. Should analysis fail to show the resonant frequency to be significantly greater than 33 hertz, a test is performed to establish the motor resonant frequency. Motor operability during a seismic event is demonstrated by calculating critical deflections, loads, and stresses under various combinations of seismic, gravitational, and operational loads. The worst case (maximum) values calculated are tabulated against the allowable values. On combining these stresses, the most unfavorable possibilities are considered in the following areas:

1. Maximum rotor deflection.
2. Maximum shaft stresses.
3. Maximum bearing load and shaft slope at the bearings.
4. Maximum stresses in the stator core welds.
5. Maximum stresses in the stator core to frame welds.
6. Maximum stresses in the motor mounting bolts.
7. Maximum stresses in the motor feet.

Where minor differences exist between items of equipment, analysis is employed to demonstrate that the test results obtained for one piece of equipment are equally applicable to a similar piece of equipment (see Table 3.10-1, EQDP ESE-23 and ESE-25).

The analytical models employed and the results of the analysis are described in Section 4 of the applicable EQDPs (Reference 2).

3.10.2.3 Combined Analysis and Testing

When the equipment cannot be practically qualified by analysis or testing alone because of its size and complexity, combined analysis and testing is utilized. One of the methods described in Sections 7.2 through 7.5 of IEEE Standard 344-1975 is used when this type of qualification is necessary.

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

Where supports for the electrical equipment and instrumentation are within the NPB scope, the seismic qualification tests and/or analysis are conducted including the supplied supports. The EQDPs contained in Reference 2 identify the equipment mounting employed for qualification purposes and establish interface requirements for the equipment to ensure that subsequent in-plant installation does not prejudice the qualification established by Westinghouse.

Analysis or tests are performed for all supports of BOP electrical equipment and instrumentation such as battery racks, instrument racks, control consoles, cabinets, panels and cable trays to ensure their structural capability to withstand seismic excitation.

3.10.4 Operating License Review

The results of the tests and analyses to demonstrate adequate seismic qualification and implementation of proper criteria for NSSS items will be presented in the RESAR-SP/90 FDA version.

3.10.5 References

1. Butterworth, G. and Miller, R. B., "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," WCAP-8587, Revision 6, November 1983.

2. "Equipment Qualification Data Packages," WCAP-8587, Supplement 1, latest revision.

3. Morrone, A. "Seismic Vibration Testing with Sine Beats," WCAP-7558, October, 1971.

4. Letter NS-CE-692, dated July 10, 1975, C. Eicheldinger (Westinghouse) to D. B. Vassallo (NRC).

5. Jarecki, S. J., "General Method of Developing Multifrequency Biaxial Test Inputs for Bistables," WCAP-8624 (Proprietary), September, 1975 and WCAP-8695 (Non-Proprietary), August, 1975.

TABLE 3.10-1 (Page 1 of 3)

SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Equipment	Equipment Qualification Data Package Reference*
Safety related valve electric motor-operators	HE-1 and HE-4
Safety related pilot solenoid valves	HE-2 and HE-5
Safety related externally mounted limit switches	HE-3 and HE-6
Medium pump motors (outside containment)	AE-1
Large pump motors (outside containment)	AE-2
Canned pump motors (outside containment)	AE-3
Pressure transmitters	ESE-1 and ESE-2
Differential pressure transmitters	ESE-3 and ESE-4
Resistance temperature detectors	ESE-6 and ESE-7
Main control board switch modules	ESE-12
Indicators (post-accident monitoring)	ESE-14
Recorders (post-accident monitoring)	ESE-15
Containment pressure sensor	ESE-21
Four section excore neutron detector	ESE-22
Reactor coolant pump speed sensor	ESE-24
Main control board Primary control console Secondary control console Safety center	ESE-25
Reactor trip switchgear	ESE-26
Nitrogen-16 detector	ESE-27
Rod position detector	ESE-28

* Refer to WCAP-8587, Supplement 1 (Reference 2).
** Items listed as "Later" will be addressed in plant specific applicant's FDA.

TABLE 3.10-1 (Page 2 of 3)

SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Equipment	Equipment Qualification Data Package Reference*
Rod position data cabinet	ESE-29
Integrated protection cabinet	ESE-30
Integrated logic cabinet	ESE-31
Field termination cabinet	ESE-32
Instrument bus distribution panel	ESE-33 and ESE-34
Instrument power supply (static invertor)	ESE-35
Post-accident monitoring system demultiplexer	ESE-37
Control board multiplexer	ESE-38
Fiber optic cable	ESE-39
Emergency diesel generator	Later**
Room coolers	Later
Safety related fans	Later
Air cleaning devices	Later
Packaged A/C units	Later
Dampers - HVAC	Later
Emergency feedwater pump turbine	Later
Electric H_2 Recombiner	WCAP-7709L
Main steam and main feedwater isolation valves	Later
Small motors	Later
Containment butterfly valves	Later

* Refer to WCAP-8587, Supplement 1 (Reference 2).
** Items listed as "Later" will be addressed in plant specific applicant's FDA.

TABLE 3.10-1 (Page 3 of 3)

SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Equipment	Equipment Qualification Data Package Reference*
Electrical distribution switchgear	Later**
Electrical penetrations	Later
Transformers	Later
Prefabricated cable assemblies	Later
Load shedder and emergency load sequencer	Later
Motor control centers	Later
AC/DC switchboards	Later
Batteries and battery racks	Later
Battery chargers	Later
Local control stations	Later
Auxiliary relay racks	Later
Main control boards	Later
Radiation monitors/airborne radioactivity monitors	Later
Control board HVAC chlorine monitor	Later

* Refer to WCAP-8587, Supplement 1 (Reference 2).

** Items listed as "Later" will be addressed in plant specific applicant's FDA.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

This section presents information to demonstrate that the mechanical and electrical portions of the engineered safety features and the reactor protection systems are capable of performing their designated safety related functions while exposed to applicable normal, abnormal, test, accident, and post-accident environmental conditions. The information presented includes identification of the safety related equipment within the Nuclear Power Block (NPB). Additionally and for each item of equipment, the designated safety related functional requirements, definition of the applicable environmental parameters, and documentation of the qualification process employed to demonstrate the required environmental capability are provided. The seismic qualification of safety related mechanical and electrical equipment is presented in Sections 3.9 and 3.10, respectively. Interface information for the environmental design of mechanical and electrical equipment is presented in Appendix 3A.

Environmental design criteria for the facilities conform to 10CFR50, Appendix A, General Design Criteria 4, Environmental and Missile Design Bases.

3.11.1 Equipment Identification and Environmental Conditions

A complete list of safety related equipment that is required to function during and subsequent to an accident is presented in Table 3.11-1. This list includes appropriate items within the NPB. Items in the NPB scope include; (1) the containment building, (2) the fuel handling facilities, (3) the mechanical safeguards equipment area, (4) the auxiliary systems area, (5) the instrumentation and controls area, (6) the control room, (7) the electrical power distribution equipment area, (8) the emergency diesel generators area, and (9) the technical support center. A detailed list of NPB scope items can be found in Table 1.1-1.

The environmental parameters employed by Westinghouse for generic qualification purposes are described in Reference 1 and specified in Reference 2 as applied to the individual equipment qualification programs.

3.11.2 Qualification Tests and Analyses

3.11.2.1 Environmental Qualification Criteria

The methods of meeting the general requirements for environmental design and qualification of safety related equipment as described by General Design Criteria 1, 2, 4, and 23 are described in Section 3.1. Additional specific information concerning the implementation of General Design Criteria 23 is provided in Section 7.2 of RESAR-SP/90 PDA Module 9, "I&C and Electrical Power." The general methods of implementing the requirements of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," are described in Chapter 17.0.

Westinghouse will meet IEEE Standard 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," including IEEE Standard 323a-1975 (the Nuclear Power Engineering Committee (NPEC) Position Statement of July 24, 1975) by either type test, operating experience, analysis, or an appropriate combination of these methods. Westinghouse will meet this commitment employing the methodology described in the final NRC approved version of Reference 1.

3.11.2.2 Performance Requirements for Environmental Qualification

In response to the NRC staff request for additional detailed information on the qualification program, Westinghouse submitted supplement 1 to WCAP-8587. The latest revision of this supplement, Reference 2, contains an equipment qualification data package (EQDP) for every item of safety related electrical equipment supplied by Westinghouse within the nuclear power block scope of supply. Table 3.11-1 identifies the equipment supplied and identifies the applicable EQDP contained in Supplement 1.

Each EQDP in Supplement 1 contains a section entitled "Performance Specification." This specification establishes the safety related functional requirements of the equipment to be demonstrated under normal, abnormal, test, accident, and post-accident conditions. The environmental qualification parameters (e.g., temperature, humidity, pressure, radiation, etc.) to be employed by Westinghouse for generic qualification purposes are also identified in the specification as applicable.

3.11.2.3 Methods and Procedures for Environmental Qualification

The basic methodology to be employed by Westinghouse for qualification of safety related electrical equipment is described in Reference 1. Each EQDP (Reference 2) contains a description of the qualification program plan for that piece of equipment. Qualification may be demonstrated by either type test, operating experience, analysis, or a combination of these methods.

3.11.3 Qualification Test Results

Qualification program results will be provided in the RESAR-SP/90 FDA version.

3.11.4 Loss of Ventilation

Refer to the plant specific applicant's safety analysis report for a discussion of loss of ventilation.

3.11.5 Estimated Chemical and Radiation Environment

Generic estimates of the radiation dose incurred by equipment during normal operation are provided in Reference 1. The estimated doses and chemical conditions following an accident are defined in Reference 1 and specified in Reference 2 as they apply to the individual equipment qualification program plans.

3.11.6 References

1. Butterworth, G. and Miller, R. B., "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," WCAP-8587, Revision 6, November 1983.

2. "Equipment Qualification Data Packages," WCAP-8587, Supplement 1, latest revision.

TABLE 3.11-1 (Page 1 of 3)

SAFETY RELATED EQUIPMENT

Equipment	Equipment Qualification Data Package Reference*
Safety related valve electric motor-operators	HE-1 and HE-4
Safety related pilot solenoid valves	HE-2 and HE-5
Safety related externally mounted limit switches	HE-3 and HE-6
Medium pump motors (outside containment)	AE-1
Large pump motors (outside containment)	AE-2
Canned pump motors (outside containment)	AE-3
Pressure transmitters	ESE-1 and ESE-2
Differential pressure transmitters	ESE-3 and ESE-4
Resistance temperature detectors	ESE-6 and ESE-7
Excore neutron detectors	ESE-9
Main control board switch modules	ESE-12
Indicators (post-accident monitoring)	ESE-14
Recorders (post-accident monitoring)	ESE-15
Containment pressure sensor	ESE-21
Four section excore neutron detector	ESE-22
Reactor coolant pump speed sensor	ESE-24
Main control board Primary control console Secondary control console Safety center	ESE-25
Reactor trip switchgear	ESE-26
Nitrogen-16 detector ESE-27	

* Refer to WCAP-8587, Supplement 1 (Reference 2).
** Items listed as "Later" will be addressed in plant specific applicant's FDA.

TABLE 3.11-1 (Page 2 of 3)
SAFETY RELATED EQUIPMENT

Equipment	Equipment Qualification Data Package Reference*
Rod position detector	ESE-28
Rod position data cabinet	ESE-29
Integrated protection cabinet	ESE-30
Integrated logic cabinet	ESE-31
Field termination cabinet	ESE-32
Instrument bus distribution panel	ESE-33 and ESE-34
Instrument power supply (static invertor)	ESE-35
Source range preamplifier	ESE-36
Post-accident monitoring system demultiplexer	ESE-37
Control board multiplexer	ESE-38
Fiber optic cable	ESE-39
Emergency diesel generator	Later**
Room coolers	Later
Safety related fans	Later
Air cleaning devices	Later
Packaged A/C units	Later
Dampers - HVAC	Later
Emergency feedwater pump turbine	Later
Electric H_2 Recombiner	Later
Main steam and main feedwater isolation valves	Later

* Refer to WCAP-8587, Supplement 1 (Reference 2).
** Items listed as "Later" will be addressed in plant specific applicant's FDA.

TABLE 3.11-1 (Page 3 of 3)

SAFETY RELATED EQUIPMENT

Equipment	Equipment Qualification Data Package Reference*
Small motors	Later
Containment butterfly valves	Later
Electrical distribution switchgear	Later
Electrical penetrations	Later
Transformers	Later**
Prefabricated cable assemblies	Later
Load shedder and emergency load sequencer	Later
Motor control centers	Later
AC/DC switchboards	Later
Batteries and battery racks	Later
Battery chargers	Later
Local control stations	Later
Auxiliary relay racks	Later
Main control boards	Later
Radiation monitors/airborne radioactivity monitors	Later
Control board HVAC chlorine monitor	Later

* Refer to WCAP-8587, Supplement 1 (Reference 2).
** Items listed as "Later" will be addressed in plant specific applicant's FDA.